Internal Dosimetry for Uranium:

General Concepts
Dose Parameters & Limits
Biokinetic Models
Bioassay Methods
Intake Estimates

Module Objectives

- Describe the primary and derived limits for internal dose in 10CFR20
- Be familiar with the standard ICRP 30 models.
- Be familiar with the newer ICRP models (ICRP Publications 60, 66, 68, 72, 78, and 103).
- State acceptable procedures and relative merits of whole-body counting for intake assessment.
- Be familiar with the various approved methods for intake assessment and dose calculation.

Module Objectives (cont.)

- Be familiar with the acceptable methods for bioassay scheduling and record keeping, reporting requirements, and quality assurance in an internal dosimetry program.
- Be familiar with the NRC regulatory guidance on internal dosimetry.

Types of Radiation Effects

- Stochastic effect: effect whose severity is NOT dependent on the dose received, but the probability of occurrence does depend on the dose. It is assumed that there is no threshold: the probability of occurrence drops to zero only at zero dose. Examples are cancer and genetic effects.
- Non-Stochastic effect: also known as a deterministic effect, its severity is proportional to the dose. Non-stochastic effects exhibit a threshold: there is no effect below a certain dose. Examples are radiation burns, acute radiation syndrome, etc.

Types of Radiation Effects, con't

- Somatic effects: occur in the person who was irradiated, i.e., cancer induction (stochastic) or non-stochastic (deterministic) effects
- Genetic effects: occur in the progeny of the person who was irradiated, i.e., lethal mutations in the next two generations

Dose Equivalent (H_T)

- The product of the absorbed dose in tissue T, the quality factor (or radiation weighting factor), and all other necessary modifying factors at the location of interest.
- The unit of dose equivalent is the rem or sievert (Sv)

  (1 Sv = 100 rem)
- This quantity depends on the amount of activity in the organ and elsewhere in the body, the type of radiation, the energy released per nuclear transformation, the fraction of the energy absorbed, the mass of the organ, and the time period over which we integrate.

Committed Dose Equivalent ($H_{50,T}$)

- Radioactive materials can be retained in the body for long times, and therefore continue to irradiate the organs.
- The CDE is the dose equivalent to a given organ or tissue that will be received in the 50 year period following an intake of radioactive material.
- Although the CDE is calculated for 50 years post intake, it is all assigned to the year of intake for dose limitation purposes .
- Unit of CDE is rem or Sv.
- CDE is limited to preclude non-stochastic effects.

Total Organ Dose Equivalent ($H_{50,T}$)

- The TODE is the sum of the CDE and the deep dose equivalent (DDE) from external penetrating radiation.
- TODE = DDE + CDE
- Unit is rem or Sv
- On an annual basis the TODE received from all external radiation and all intakes within a year must be less than 50 rem (0.5 Sv).
- The TODE limit is a non-stochastic dose limit.

Committed Effective Dose Equivalent (E_{50})

- The CEDE was developed to deal with the fact that radioactive materials are not uniformly distributed in the body, and so different organs may receive different CDEs
- The CEDE is the weighted sum of the products of the tissue weighting factors (w_T) applicable to each of the body organs or tissues that are irradiated, and the CDE to each of those organs or tissues.
- CEDE = $\Sigma\ w_T\ H_{50,T}$
- The CEDE is considered a stochastic dose limit.

Weighting Factor w_T

- The tissue weighting factor relates the stochastic risk from irradiation of one organ or tissue to the total stochastic risk when the whole body is (externally) irradiated uniformly.
- Put another way, if a large population were all irradiated, the tissue weighting factor gives the fraction of all the cancers that would occur in a given organ.
- The weighting factors from ICRP 26 are given in 10CFR20.
- Revised weighting factors are given in ICRP 60 and 103

ICRP 26 TISSUE WEIGHTING FACTORS[1]

Organ or tissue	W_T
Gonads	0.25
Breast	0.15
Red bone marrow	0.12
Lung	0.12
Thyroid	0.03
Bone surfaces	0.03
Remainder	0.30[2]
Whole Body	1.00[3]

[1] required by 10CFR20

[2] 0.30 results from 0.06 for each of 5 "remainder" organs (excluding the skin and the lens of the eye) that receive the highest doses.

[3] For the purpose of weighting the external whole body dose (for adding it to the internal dose), a single weighting factor, w_T=1.0, has been specified. The use of other weighting factors for external exposure will be approved on a case-by case basis until such time as specific guidance is issued.

Total Effective Dose Equivalent

- The TEDE is the sum of the DDE for external exposures and the CEDE for all intakes in the year.
- The TEDE is the reported “whole body” dose.
- The TEDE is a stochastic limit and the regulatory limit is 5 rem (50 mSv) per year.

NRC Limits

- From a regulatory standpoint, NEITHER the stochastic nor the non-stochastic limit may be exceeded; that is, the TEDE must be no greater than 5 rem and the TODE for any organ or tissue must be no more than 50 rem.
- TODE < 50 rem (500 mSv) per year
- TEDE < 5 rem (50 mSv) per year

Important Definitions

- INTAKE is the total amount of radioactive material that entered the body.
- UPTAKE is the fraction of the intake that is absorbed into the extracellular fluids (mostly the blood and lymph, which are stream also called the transfer compartment.)
- DEPOSITION is the sequestration of radioactive material from the transfer compartment into a specific organ.
  - "Initial deposition" refers to the fraction of the intake retained at the intake site, e.g., in the lungs.

Intake Routes

There are four possible routes of intake (in order of likelihood):

- Inhalation: breathing in radioactive material
- Ingestion: taking radioactive material into the gastrointestinal tract, usually by swallowing
  - some inhaled materials enter the GI tract through mucociliary clearance from the lung, and some enter via hepatobiliary excretion from the liver, but this is not considered ingestion.
- Injection: a cut or puncture wound from a contaminated object.
- Direct (percutaneous) absorption through the skin: limited to a few nuclides in specific chemical forms, e.g., HTO.

Elimination Pathways

- Just as there are four intake pathways there are four elimination pathways from the body.
  - Urine
  - Feces
  - Breath (for gaseous radionuclides either inhaled or produced in the body, e.g., $^{14}CO_2$
  - Perspiration
- For uranium compounds urine and fecal are the elimination pathways.

Elimination of Radioactive Material

- Clearance of radionuclides from the body occurs via the elimination pathways and from radioactive decay.
- Both of these processes occur with a defined elimination constant.
- The biological elimination constant (λ_B) and the radioactive decay constant (λ_R) can be used to calculate the effective elimination constant (λ_E).
- The equation for this is $\lambda_E = \lambda_R + \lambda_B$

Effective Half Life

- The elimination constants can be used to calculate the half life for each component of the equation:

$$T = 0.693/\lambda$$

- An equation for the effective half life can then be determined.

$$T_E = \frac{T_R \times T_B}{T_R + T_B}$$

Effective Half Life of Uranium

- Since all the uranium isotopes of interest have very long half lives, radioactive decay is negligible compared to excretion.
- The effective half life essentially equals the biological half life.
- For any radionuclide, the effective half life is always less than the smaller of the radiological and biological half lives.

Effective Half Life Example

- There are different biological half lives based on where the uranium is at in the body. There is one for the transfer from the lung to blood, blood to specific organs, or from specific organ to excretion.
- In this example we will compute the effective half life of U-238 in the bone.
- The radioactive half life is 4.468 X 10^9 years (1.632 X 10^{12} days) and the biological half life in the bone is 1500 days.

$$T_E = \frac{1.632^{12} days \times 1500\ days}{1.632^{12} days + 1500\ days} = 1500 days$$

ICRP 30 Models

- International Commission on Radiation Protection (ICRP) Publication 30 (1979) details the models to use for calculation of internal doses.
- It replaced ICRP Publication 2 (1959), which specified maximum permissible concentrations in air and water based on maximum permissible body burdens.
- Current NRC regulations (10CFR20) are based on ICRP 30.
- The US Department of Energy regulations (10CFR835) are transitioning to the newer models in ICRP Publication 60 (1990) and following

ICRP 30 Models

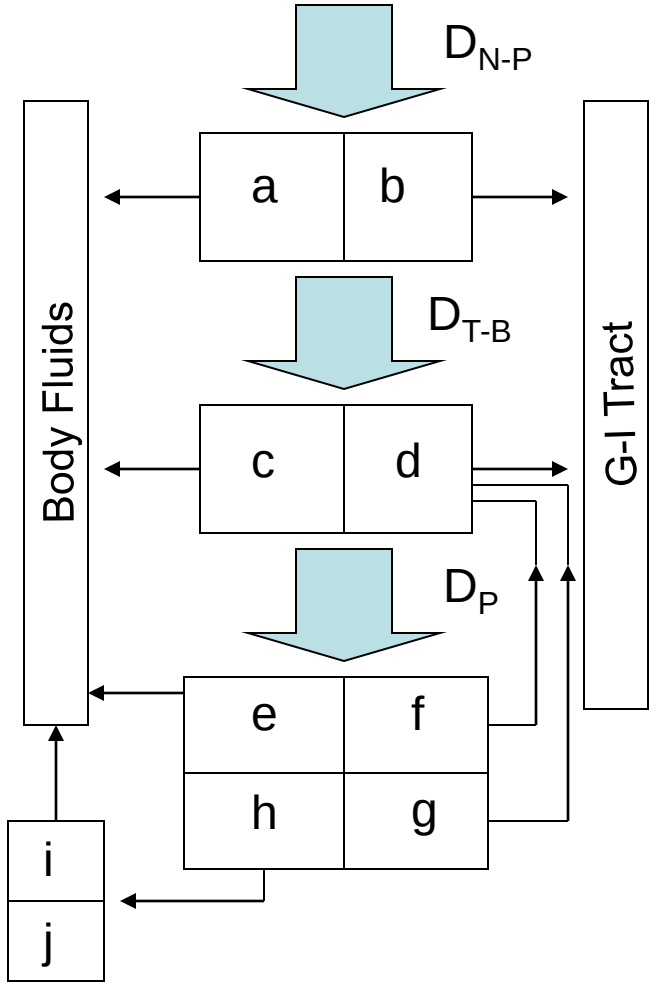
- The ICRP 30 models address the intake, distribution, retention and excretion of radionuclides from the body.
- For uranium the main routes of intake are inhalation and ingestion.
- There are three ICRP 30 models that we are concerned with.
  - Lung clearance model
  - Gastrointestinal tract model
  - Bone deposition model

Lung Clearance Model

- The ICRP 30 lung model is composed of regions that are anatomical and compartments that refer to clearance mechanisms.
- The respiratory tract is divided into three regions
  - Nasal Passages (NP)
  - Trachea and Bronchi (TB)
  - Pulmonary Parenchyma (P)
    - Pulmonary lymphatic system (L)
- The lung model ignores doses to the NP region; the “lung” consists of the TB, P, and L regions, with a mass of 1 kg.

ICRP 30 Lung Model

Region	Compartment	Class					
		D		W		Y	
		T	F	T	F	T	F
N-P	a	0.01	0.5	0.01	0.1	0.01	0.01
(D_{N-P} = 0.30)	b	0.01	0.5	0.40	0.9	0.40	0.99
T-B	c	0.01	0.95	0.01	0.5	0.01	0.01
(D_{T-B} = 0.08)	d	0.2	0.05	0.2	0.5	0.2	0.99
P	e	0.5	0.8	50	0.15	500	0.05
(D_P = 0.25)	f	n.a.	n.a.	1.0	0.4	1.0	0.4
	g	n.a.	n.a.	50	0.4	500	0.4
	h	0.5	0.2	50	0.05	500	0.15
L	i	0.5	1.0	50	1.0	1000	0.9
	j	n.a.	n.a	n.a	n.a.	∞	0.1



Task Group Lung Model (ICRP 30)

- There are two primary clearance pathways and one minor clearance pathway in the lung:
  - Direct absorption into body fluids (the transfer compartment) from compartments a, c, e, and i.
  - Mechanical clearance via mucociliary action up the bronchi and trachea to the throat and gastrointestinal tract represented by compartments b, d, f, and g.
  - Biological clearance to the pulmonary lymph nodes, primarily due to ingestion by macrophages in compartment h and permanent retention in compartment j

Lung Clearance Model

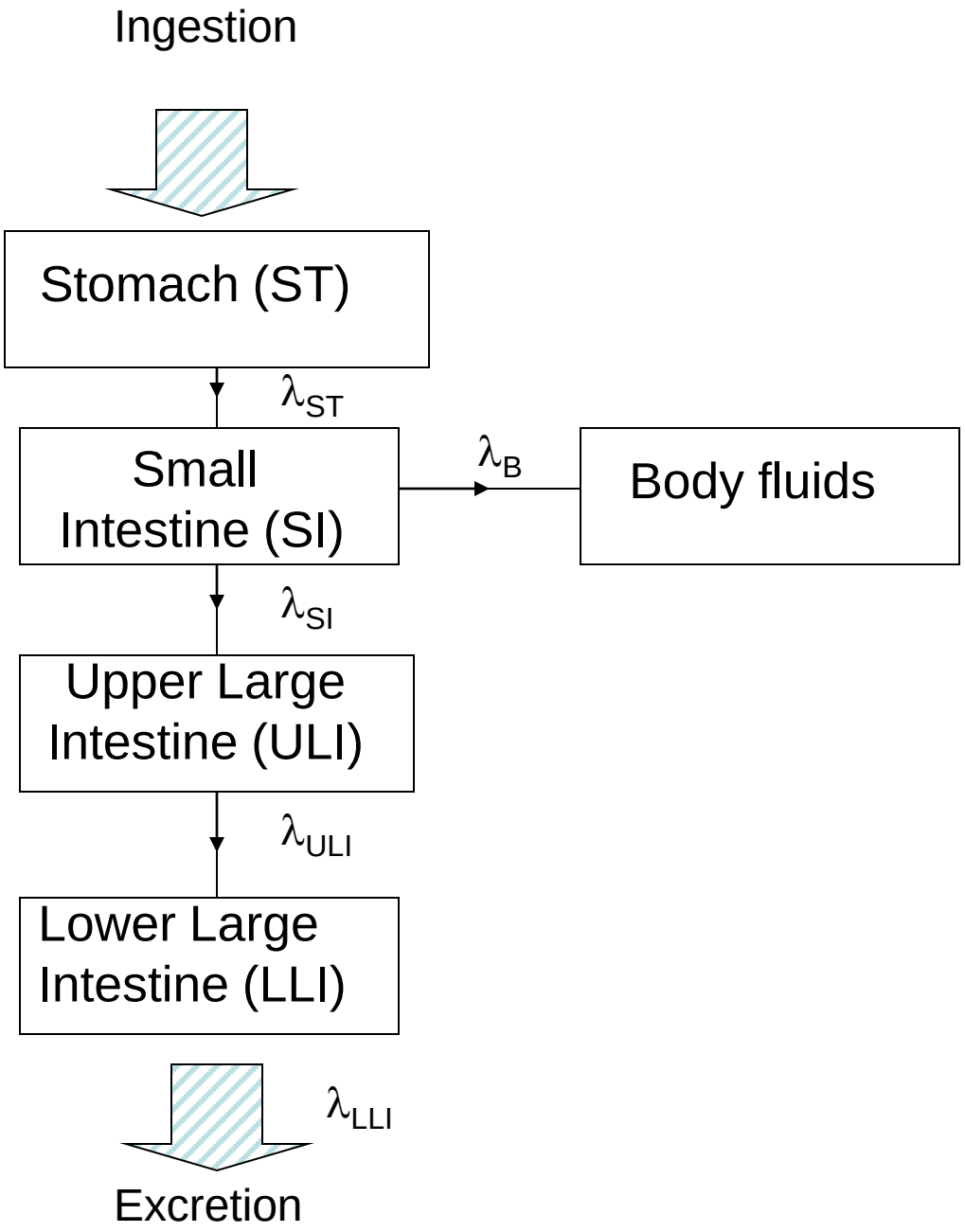
- Inhaled materials deposit in each region as a function of particle size.
- The default particle size for the ICRP 30 model is 1 micron AMAD (activity median aerodynamic diameter).
- Initial deposition is 30% in NP, 8% in TB and 25% in P
- Radionuclides are assigned to one of three solubility classes:
  - Class D clears the lung with a half-time of days (default value is 0.5 days)
  - Class W clears the lung with a half-time of weeks (default value is 50 days)
  - Class Y clears the lung with a half-time of years (default value is 500 days).

Uranium Inhalation Classes

- The inhalation class of each radionuclide as a function of its chemical form is given in Federal Guidance Report 11, Table 3 (as well as 10CFR20, App. B):
- Class D: UF_6, UO_2F_2, $UO_2(NO_3)_2$
- Class W: UO_3, UF_4, UCl_4
- Class Y: UO_2, U_3O_8
- DOE HP Manual for Good Practices for Uranium Facilities (EGG-2350, 1998) includes more compounds, but disagrees with FGR-11 in some cases

Gastrointestinal Tract

- The ICRP 30 gastrointestinal tract model is somewhat less complicated than the lung model.
- The GI tract is divided into four sequential regions:
  - Stomach
  - Small intestine
  - Upper large intestine
  - Lower large intestine
- Absorption from the GI tract into the body fluids is assumed to occur only from the small intestine.


Ingestion
Stomach (ST)
λ_{ST}
Small Intestine (SI)
λ_{B}
Body fluids
λ_{SI}
Upper Large Intestine (ULI)
λ_{ULI}
Lower Large Intestine (LLI)
λ_{LLI}
Excretion

Gastrointestinal Tract Model

- The lambda values represent the clearance time constant from one compartment to the next.
- The most important factor is the f_1 value, which is the fraction of radioactive material absorbed from the small intestine into the body fluids (transfer compartment).
- The f_1 values for uranium are:
  - Class D 0.05
  - Class W 0.002
  - Class Y 0.002

Bone Dosimetry Model

- The third ICRP 30 model of interest is the bone model, which has two types of bone:
  - Cortical bone, found in the midshafts of the long bones
  - Trabecular bone, with a honeycomb appearance, found at the ends of the long bones and in the flat bones.
- Each type contains one half of the bone surface but cortical bone has 80% of the mass and trabecular bone has 20% (skeletal mass 7 kg, mineral mass 5 kg)
- For dose calculations, there are two tissues: bone surfaces (BS) and red marrow (RM), found only in trabecular bone.

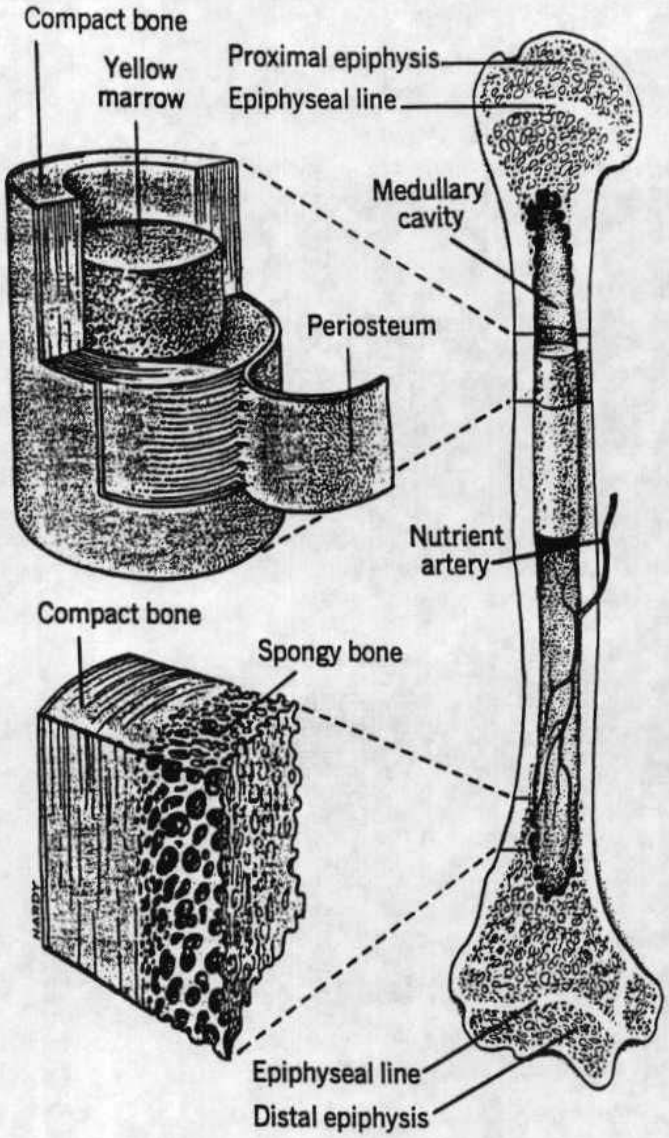


Figure 54-2 A long bone shown in longitudinal section. (*Chaffee EE, Lytle IM: Basic Physiology and Anatomy, 4th ed. Philadelphia, JB Lippincott, 1980*)

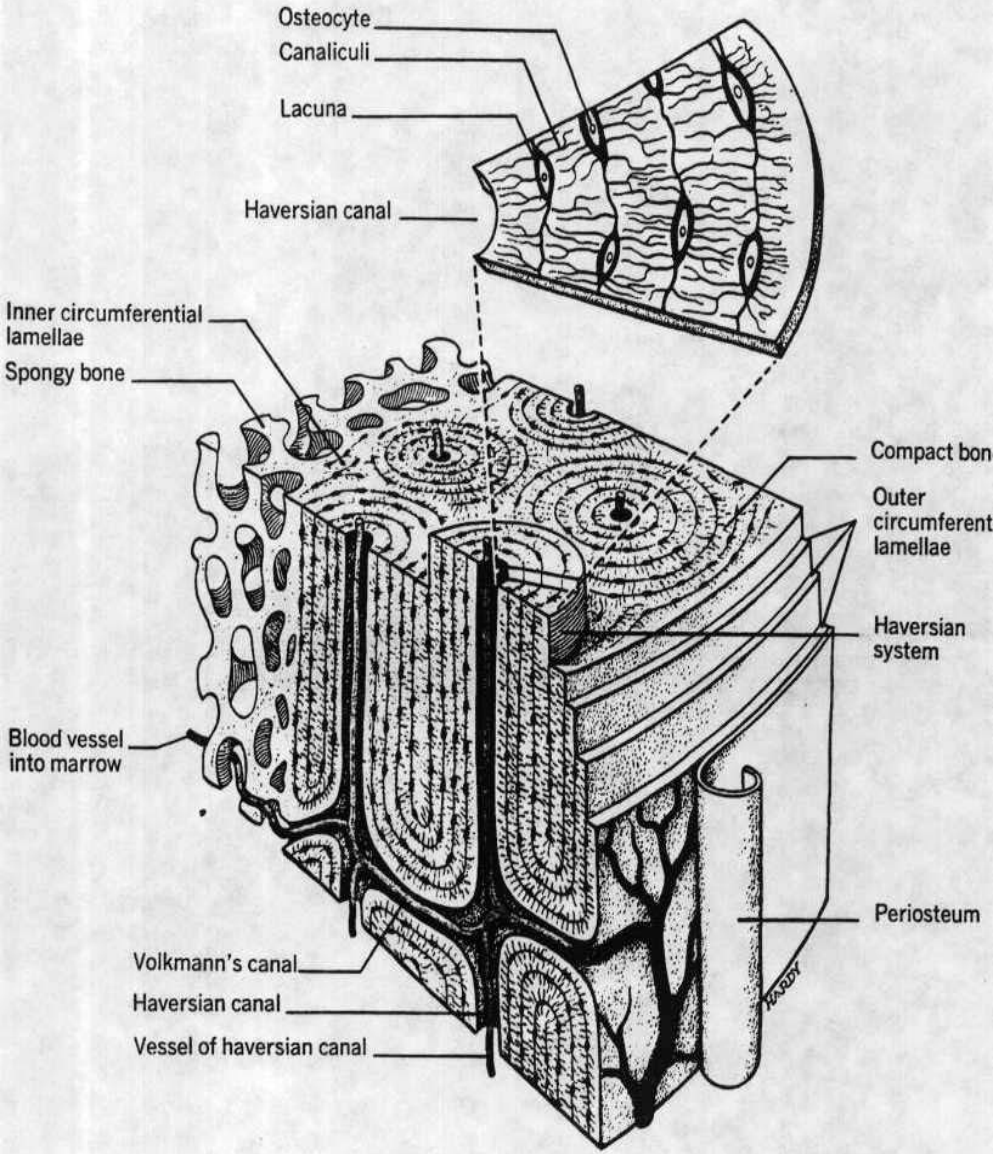


Figure 54-4 Haversian systems as seen in a wedge of compact bone tissue. The periosteum has been peeled back to show a blood vessel entering one of Volkmann's canals. (*Upper right*) Osteocytes lying within lacunae; canaliculi permit interstitial fluid to reach each lacuna. (*Chaffee EE, Lytle IM: Basic Physiology and Anatomy, 4th ed. Philadelphia, JB Lippincott, 1980*)

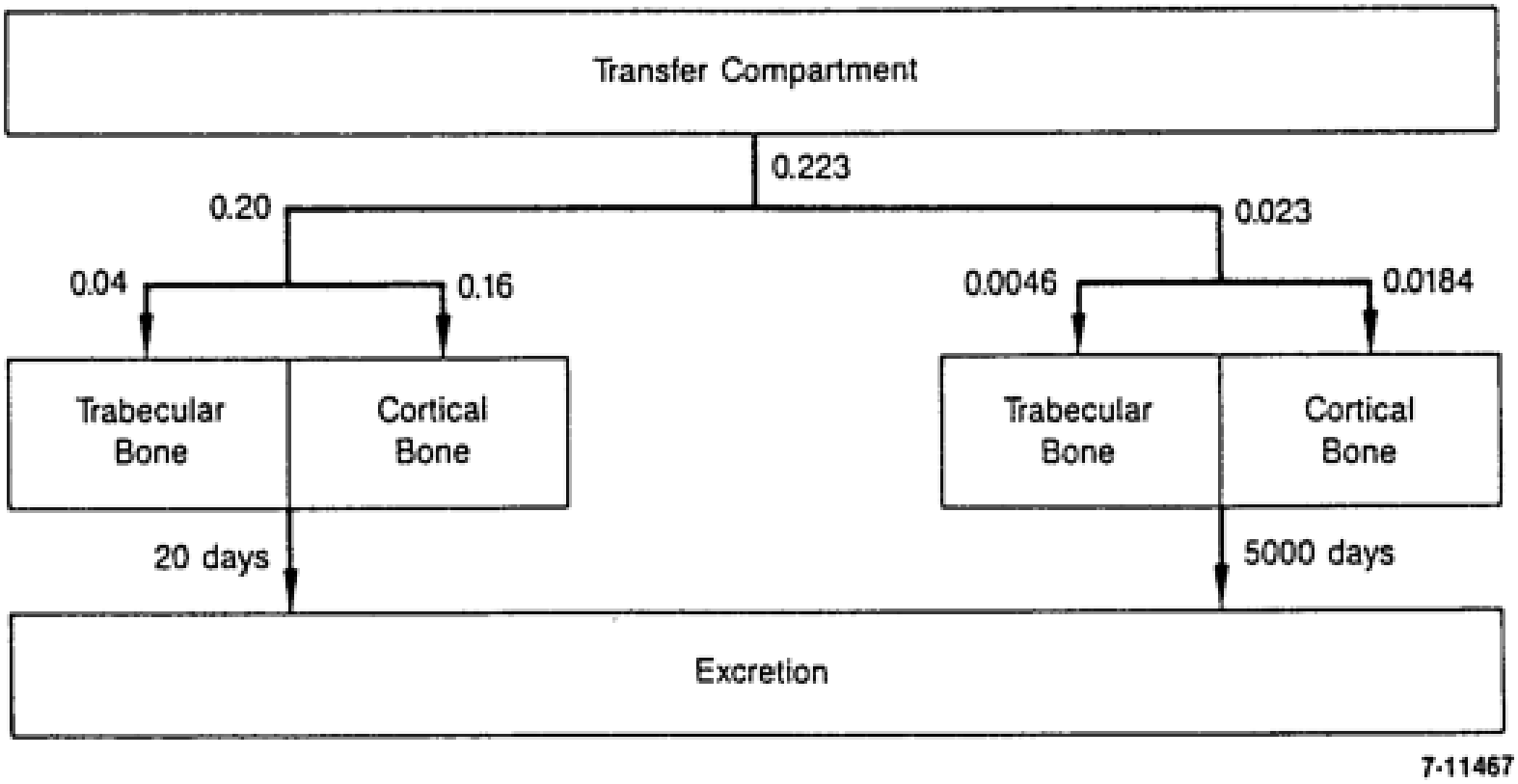
Bone Dosimetry Model

- Radiations are divided into six classes depending on the type of radiation and the location from which it is emitted.
  - All photon emitters in body
  - Alpha emitters in bone volume
  - Alpha emitters on bone surfaces
  - Beta emitters in bone volume
  - Beta emitters on bone surfaces with E > 0.2 MeV
  - Beta emitters on bone surfaces with E < 0.2 MeV

Bone Dosimetry Model

- ICRP assumes the following distributions:
  - Actinides (U, Pu, Am, etc.) are surface seekers.
  - Alkaline earths with half-lives < 15 days are surface seekers
  - Alkaline earths with half-lives > 15 days are volume seekers
  - All radionuclides are initially deposited on bone surfaces, but the longer-lived alkaline earths (calcium analogues: Sr, Ba, Ra) diffuse into bone volume.
  - Surface seekers are assumed to stay on bone surfaces, even though in reality they are buried by the apposition of new bone.

Biokinetic Model for Uranium in Bone

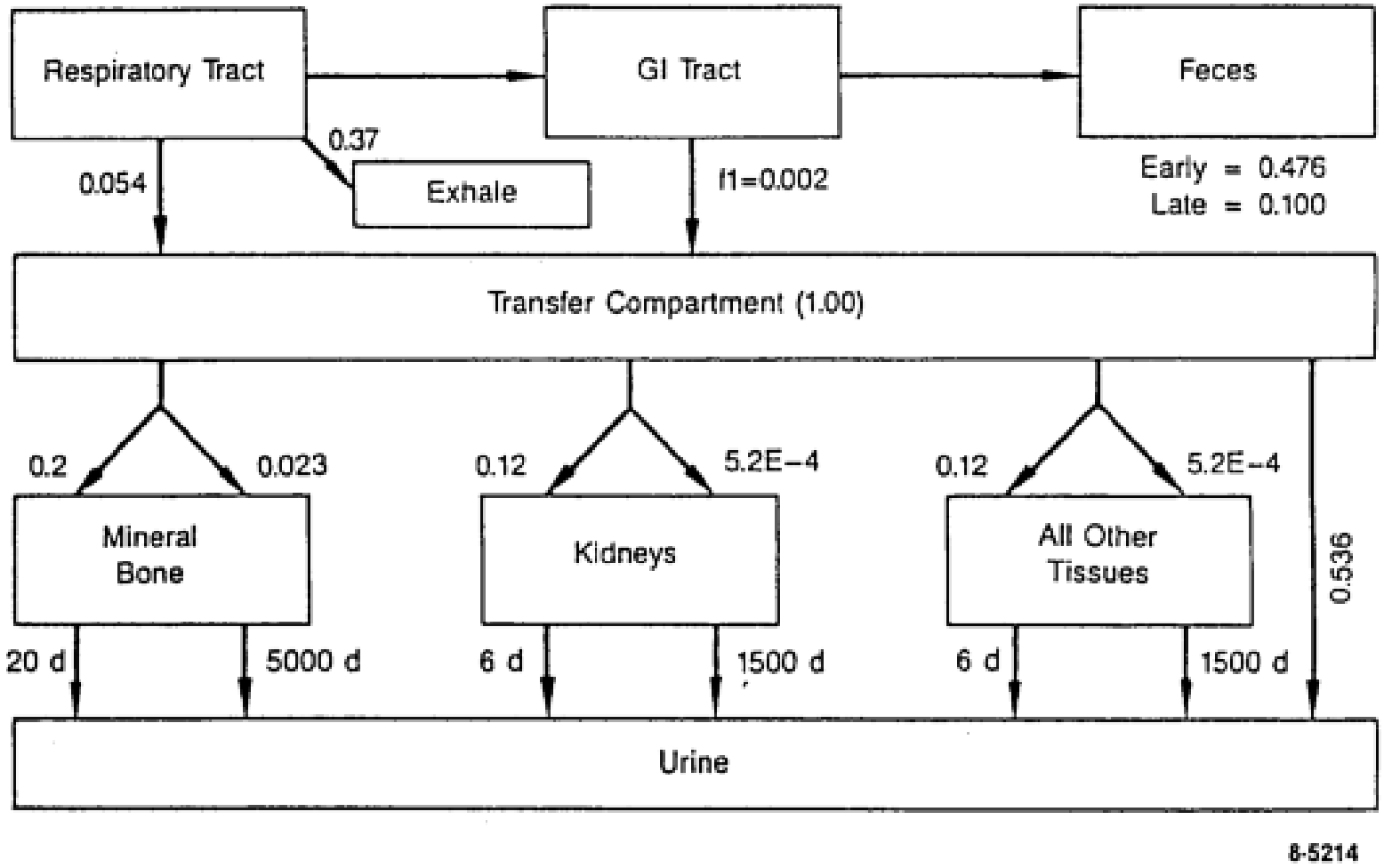


Systemic Biokinetic Model for Uranium

- The ICRP 30 systemic biokinetic model for class Y uranium is presented on the next slide.
- The fractions that reach the transfer compartment for Class D and W uranium will be different, but the biokinetics after it reaches the transfer compartment are the same.

Radionuclide = U-238 Class Y

Chemical Form = UO_2, U_3O_8



Uranium Biokinetics

- Of the uranium that reaches the transfer compartment:
  - 53.6% is promptly excreted in the urine.
  - 12% is retained in the kidney with a 6 d half-time
  - 0.052% is retained by the kidney with a 1500 d half-time.
  - 20% is deposited in bone with a 20 day retention half-time
  - 2.23% is deposited in bone with a 5000 day retention half-time
  - 12% is assumed to deposit in other soft tissues with a 6 day half-time and 0.052% with a 1500 day half time.

Radiological Toxicity

- The dose to the bone surface is limiting for inhalation and ingestion of class D uranium
- The effective dose is the limiting dose for inhalation of class W and class Y uranium
- However, for soluble compounds (class D) of depleted, natural, and low-enriched uranium, the chemical toxicity to the kidney is limiting; i.e., an intake will exceed the TLV for kidney damage before it exceeds the radiological dose limit.
- For enrichment levels above 15%, the radiological dose limit is always limiting.

Acute vs Chronic Doses

- Acute intakes of uranium are relatively simple to handle: the standard retention functions are applied to the intake.
- A chronic intake can be seen as a series of small acute intakes; eventually an equilibrium body burden will be reached with the daily excretion equal to daily intake.
- To prevent kidney toxicity, an industrial hygiene limit of 10 mg soluble uranium intake per week (0.2 mg/m^3 airborne level) has been established (10CFR20 Appendix B footnote 3).
- At this level of intake the equilibrium kidney burden would be about 1 mg which is about a factor of 3 to 10 below that observed to cause toxicity.
- NOAEL = 3 µg/g kidney x 300 g (kidney mass) ~ 1 mg

Site Specific Models

- Frequently a facility (and individual workers) will work with uranium materials of different solubilities, or different particle sizes.
- This will confound dose assessments as the intake estimate will be difficult to make as it may not match the standard biokinetic models.
- Site or individual-specific models must be documented and supported by available data.
- Once a site-specific model is derived, then dose conversion factors can be computed, and ALIs and DACs can be established.

Dose Conversion Factors
also called “dose coefficients”

- ICRP and the Environmental Protection Agency have published tables of DCFs that relate the CDE and CEDE to intakes via inhalation or ingestion of various radionuclides in various chemical forms.
- The DCF is given in Sv/Bq, and must be converted to rem/μCi.
- The conversion factor is 1.0 Sv/Bq = 3.7 E6 rem/μCi.
- These are computed for each radionuclide, for its different chemical forms, and for the two intake pathways.

FGR 11

- Federal Guidance Report 11 “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion” was published by the EPA in September 1988.
- This document contains the DCFs for all radionuclides and chemical forms for the two intake pathways of inhalation (Table 2.1) and ingestion (Table 2.2).

Committed Dose Equivalent per Unit Intake (Sv/Bq) (Inhalation)

Nuclide	Class/f1	Gonad	Breast	Lung	R Marrow	B Surface	Thyroid	Remainder	Effective
U-234	D 5E-2	2.50E-08	2.50E-08	3.18E-07	6.98E-07	1.09E-05	2.50E-08	9.26E-07	7.37E-07
	W 5E-2	7.52E-09	7.52E-09	1.60E-05	2.10E-07	3.29E-06	7.52E-09	2.85E-07	2.13E-06
	Y 2E-3	2.65E-09	2.65E-09	2.98E-09	7.22E-08	1.13E-06	2.65E-09	1.06E-07	3.58E-05
U-235	D 5E-2	2.37E-08	2.38E-08	2.95E-07	6.58E-07	1.01E-05	2.37E-08	8.59E-07	6.85E-07
	W 5E-2	7.24E-09	7.33E-09	1.48E-05	1.98E-07	3.05E-06	7.22E-09	2.65E-07	1.97E-06
	Y 2E-3	2.84E-09	5.37E-09	2.76E-04	7.15E-08	1.05E-06	4.11E-09	1.02E-07	3.32E-05
U-238	D 5E-2	2.23E-08	2.23E-08	2.80E-07	6.58E-07	9.78E-06	2.22E-08	8.22E-07	6.62E-07
	W 5E-2	6.71E-09	6.74E-09	1.42E-05	1.98E-07	2.94E-06	6.71E-09	2.54E-07	1.90E-06
	Y 2E-3	2.42E-09	2.91E-09	2.66E-04	6.88E-08	1.01E-06	2.73E-09	9.61E-08	3.20E-05

Committed Dose Equivalent per Unit Intake (Sv/Bq) (Ingestion)

Nuclide	Class/f1	Gonad	Breast	Lung	R Marrow	B Surface	Thyroid	Remainder	Effective
U-234	D,W/ 5E-2	2.59E-09	2.58E-09	2.58E-09	7.21E-08	1.13E-06	2.58E-09	1.09E-07	7.66E-08
	Y/ 2E-3	1.06E-10	1.03E-10	1.06E-10	2.88E-09	4.52E-08	1.03E-10	1.77E-08	7.06E-09
U-235	D,W/ 5E-2	2.67E-09	2.49E-09	2.46E-09	6.81E-08	1.05E-06	2.45E-09	1.03E-07	7.19E-08
	Y/ 2E-3	3.34E-10	1.21E-10	1.01E-010	2.78E-09	4.20E-8	9.82E-11	1.84E-08	7.22E-09
U-238	D,W/ 5E-2	2.31E-09	2.31E-09	2.30E-09	6.80E-08	1.01E-06	2.30E-09	9.69E-08	6.88E-08
	Y/ 2E-3	1.02E-10	9.33E-11	9.22E-11	2.72E-09	4.04E-08	9.20E-11	1.61E-08	6.42E-09

LIMITING DOSE

- The dose conversion factor printed in bold type will identify the “limiting” dose.
- If the bold entry is below an organ, then that organ will receive a CDE of 50 rem from an intake smaller than that which would produce a CEDE of 5 rem.
- If the bold entry is under “Effective”, then a CEDE of 5 rem will result from an intake smaller than that which would produce a CDE of 50 rem to any organ.
- Hint: if $w_T < 0.1$, the CDE (non-stochastic) limit will apply.

Annual Limit on Intake

- Because dose calculations can be complicated, a derived limit has been developed, the ALI.
- An ALI is defined as the dose limit divided by the DCF.
- The limit may be stochastic (5 rem CEDE) in which case it is called the SALI or non-stochastic (50 rem CDE) in which case it is called the NALI.
- THE ALI will be the lesser of the SALI and the NALI
- There are different ALIs for inhalation and ingestion, and they are dependent on the chemical form, or solubility, of the radionuclide.
- ALIs are calculated to one significant figure.

NALI Example for Inhalation

What is the ALI for Class D U-238 from inhalation?

NALI = dose limit/DCF

For class D U-238 the bolded DCF is for the bone surface so this is a non-stochastic limit.

NALI = 50 rem / (9.78 E-6 Sv/Bq * 3.7 E6 rem-Bq/µCi-Sv)

NALI = 50 rem / 36.26 rem/**µCi**

NALI = 1.38 **µ**Ci

ALI = 1 **µ**Ci

ALIs are listed in 10 CFR 20 Appendix B. If the most limiting ALI is a NALI, then the target organ will be listed along with the SALI in parenthesis.

SALI Example Inhalation

What is the SALI for class D U-238?

SALI = dose limit / DCF

SALI = 5 rem / (6.62 E-7 Sv/Bq * 3.7 E6 rem/μCi)

SALI = 5 rem / 2.45 rem/μCi

SALI = 2.04 μCi

SALI = 2 μCi

So for the example of class D U-238 the limiting dose is the 50 rem dose to the bone surface (NALI = 1 μCi).

Derived Air Concentration (DAC)

- The DAC is another derived limit that is used in air sampling and internal dosimetry.
- The DAC is defined as the ALI (for inhalation) divided by the volume of air breathed by a worker during working hours for one year (2000 hours).
- The breathing rate is assumed to be 20 liters per minute which is 2,400 cubic meters per year.

DAC Example

Based on a NALI of 1 µCi for class D U-238 what is the DAC?

DAC= ALI / 2,400 cubic meters

DAC = 1.38 µCi / 2,400 cubic meters = 5.75 E-4 µCi/ cubic meter

DAC usually given in terms of µCi/cc so need to convert

5.75 E-4 µCi / cubic meter * (1 m / 100 cm)^3 = 5.75 E-10 µCi/cc

DAC = 6 E-10 µCi/cc

Notes on Calculating DAC

- If the rounded ALI of 1 µCi had been used it would have resulted in a DAC of 4 E-10 µCi/ml.
- Round to 1 significant figure after the calculation is complete.
- The list of ALIs and DACs is available in 10 CFR 20 Appendix B.
- To calculate DAC in cubic centimeters or milliliters use 2.4 E9 cc (or ml) per year as the breathing rate.

DAC-hours

- If we assume constant, standard breathing rate, we can compute intake from DAC-hour values
- DAC is based on 2000 working hours per year, so 2000 DAC-hours = 1 ALI
- Measure the air concentration, convert it to DACs, then multiply by exposure time in hours
- Correct for respiratory protection, if used
- Can then compute dose from DAC-hours, as fraction of ALI
- But remember that the ALI is based on more restrictive of stochastic or non-stochastic limit when computing dose (CEDE or CDE)

ALIs for Ingestion

- ALIs for ingestion are also calculated and listed in 10 CFR 20 Appendix B.
- The ingestion ALI is calculated using the DCFs for ingestion
- Note: the f_1 values are already contained in the DCFs
- The calculation is the same as for inhalation, there can be both a stochastic ALI and a non-stochastic ALI based on the DCFs.

NALI Ingestion Example

- What is the ALI for class D U-238 for Ingestion?

ALI = Dose limit / DCF

For class D U-238 for ingestion the limiting DCF is bone surface 1.01 E-6 Sv/Bq

1.01 E-6 Sv/Bq * 3.7 E6 Rem/ µCi / Sv/Bq = 3.737 Rem/ µCi

ALI = 50 rem / 3.737 rem/ µCi = 13.4 µCi = 10 µCi

SALI Ingestion Example

What is the SALI for class D U-238 for Ingestion?

SALI = Dose limit / DCF

For class D U-238 for ingestion the DCF for CEDE is 6.88 E-8 Sv/Bq

6.88 E-8 Sv/Bq * 3.7 E6 rem-Bq/µCi-Sv

= 0.25 rem/µCi

ALI = 5 rem / 0.25 (rem/µCi) = 20 µCi

ALIs and DACs for Uranium

Nuclide	Class/f_1	ALI (Ingestion) µCi	ALI (inhalation) µCi	DAC µCi/ml
U-234	D/0.05	10 (Bone Surface)	1 (Bone Surface)	5 E-10
U-234	W/0.05	---	0.7	3 E-10
U-234	Y/0.002	200	0.04	2 E-11
U-235	D/0.05	10 (Bone Surface)	1 (Bone Surface)	6 E-10
U-235	W/0.05	---	0.8	3 E-10
U-235	Y/0.002	200	0.04	2 E-11
U-238	D	10 (Bone Surface)	1 (Bone Surface)	6 E-10
U-238	W	---	0.8	3 E-10
U-238	Y	200	0.04	2 E-11

Things to Remember about ALIs and DACs

- The ALIs and DACs are different for inhalation and ingestion.
- The ALIs and DACs are based on the more restrictive of the stochastic or non-stochastic limits.
- The ALI and DAC will depend on the solubility, and therefore the chemical form, of a radionuclide.
- Specific values for DCFs, ALIs, and DACs for any radionuclide depend on the exact biokinetic and intake models that are used.
- The licensee must specify the models they employ and demonstrate that revised models are appropriate for their workplace.

Determining the Intake

- Since we cannot measure internal doses directly, we base our calculations on things we can measure:
- The radioactive material contained in the body or a specific organ, or being excreted from the body.
  - There are several methods that can be used.
    - Whole body counting or lung counting
  - Excreta analysis
    - Urine (most common) or Fecal
- Air Sampling

Whole Body Counting

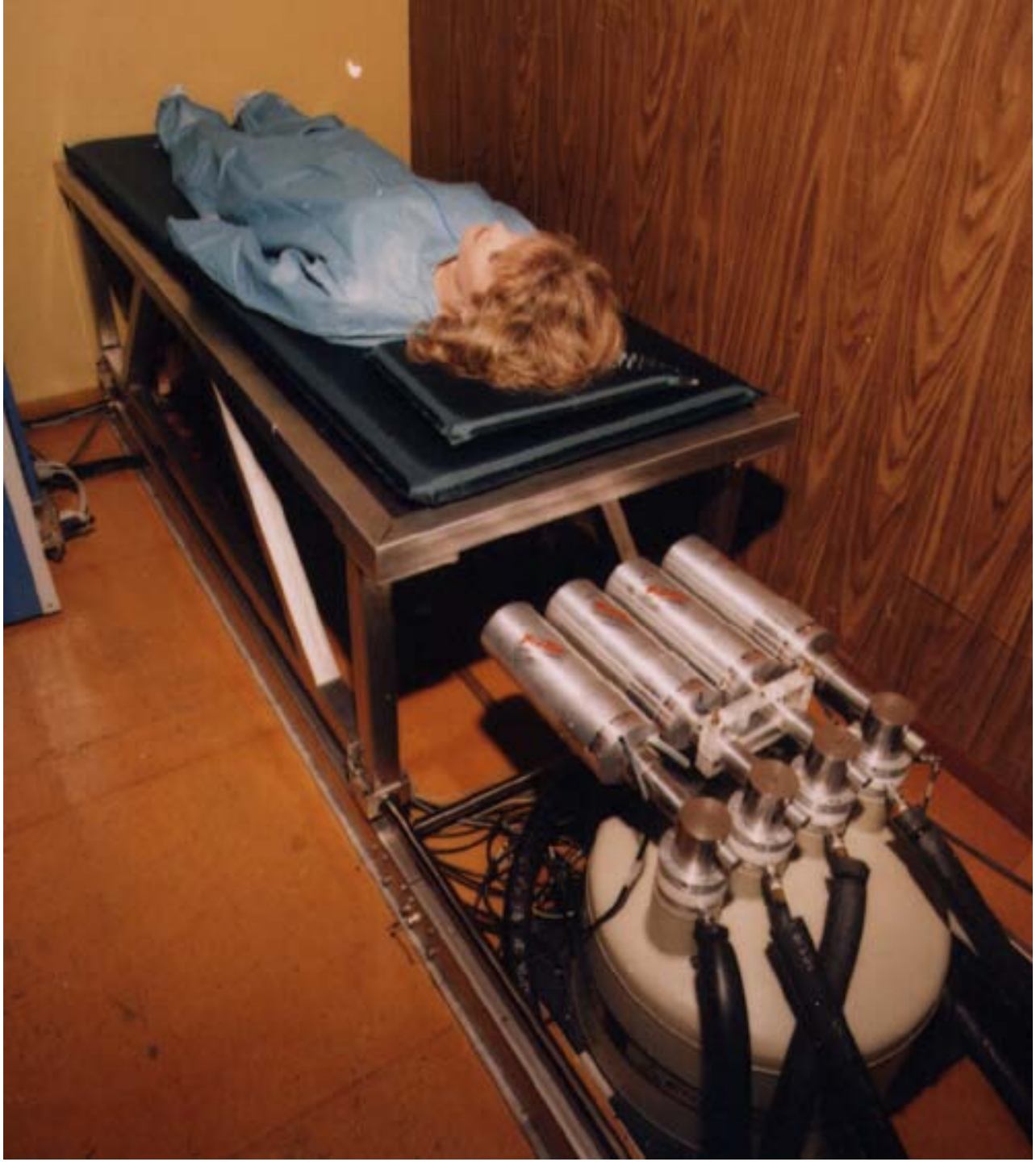
- Whole body counting (WBC) is a colloquial term for body radioactivity measurements.
- It consists of the external detection and quantification of photons that are emitted by the radioactive material in the body.
- The photons must be emitted in sufficent abundance and have high enough energy to escape the body in order to be detected.

Detectors

- There are two types of detectors that are used in whole body counting systems.
  - Thallium-activated sodium iodide [NaI(Tl)] crystals that are coupled to one or more photomultiplier tubes. These detectors have a high efficiency but poor resolution for photons that are close in energy.
  - High-purity germanium (HPGe) detectors that are smaller in size than NaI(Tl) crystals, and so have less geometric efficiency, but have much improved resolution. However, they operate at liquid nitrogen temperature.
- The HPGe detector is most commonly used for uranium measurements because of its high resolution and lower background.

WBC Setup

- WBCs can be designed in a few configurations; usually the person will lie or stand up with the detectors either stationary or moving across their body.
- WBCs are shielded in a couple of different ways.
  - One is to place the equipment and person in a shielded room.
  - Another option is to shield the detector from all sides except the one facing the person to be counted (shadow shield).
- Usually low background steel, e.g., naval armor plate from WWII ships) is used to construct shields because of contamination from melted thickness gauges (not fallout!)



Handling the Subject

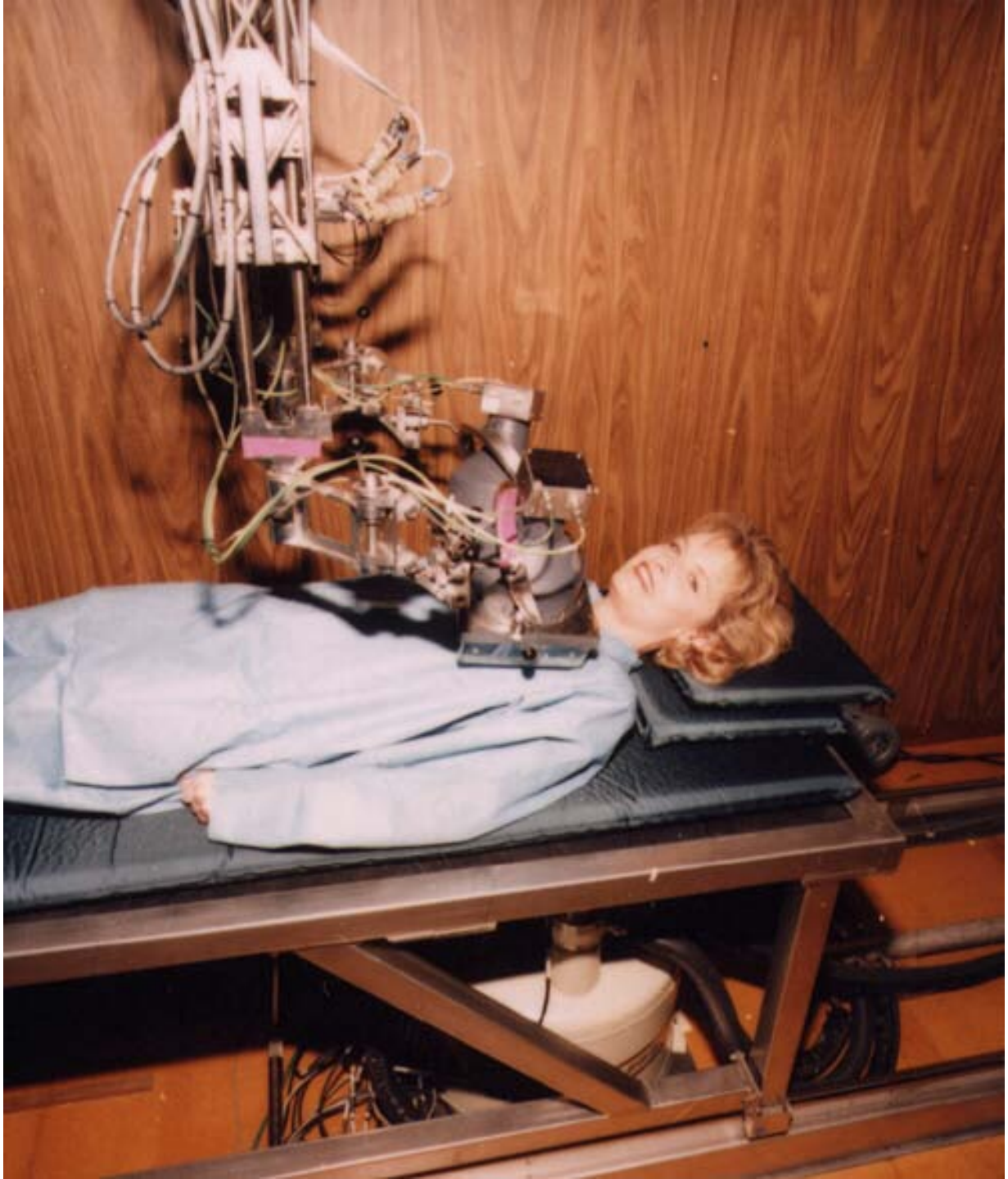
- WBCs are usually performed on a set schedule, with additional counts performed after an incident or upon completion of a special job.
- Typically the subject will leave the work area, report to the WBC, shower and change into clean clothing provided by the facility. (Omit showering if high concentrations of radon in the water!)
- Typical counting times for uranium are 30 to 40 minutes.

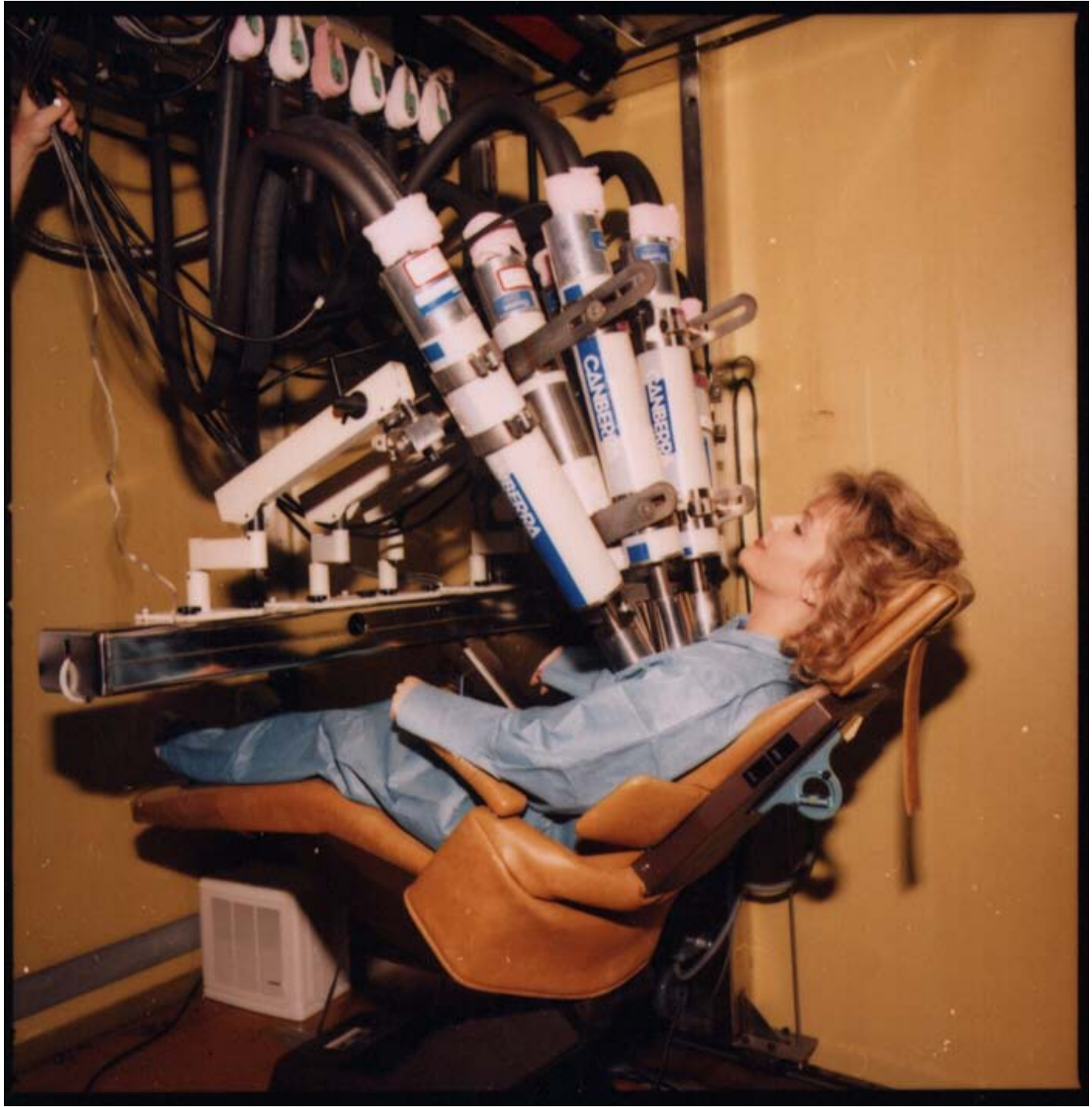
Interferences

- The biggest interference with WBC is external contamination.
- This can usually be detected by examining the spectrum from the counter and looking for the very low x-rays (10-20 keV) from uranium. These would usually not penetrate out of the body.
- Nuclear medicine procedures also are an interference.
- While most radiopharmaceuticals do not emit energies similar to those of uranium they do cause a much higher background in the counter.

Lung Counting

- Lung counts are usually performed for uranium rather than WBC.
- HPGe detectors are usually used. Most have four detectors to cover the lungs.
- The chest wall thickness must be known in order to correct the measurement for the attenuation in tissue of the low energy photons: 63 and 93 keV from Th-234, from decay of U-238 and the185-keV photon from U-235.
- The chest wall thickness can be estimated from published corrections for height, weight, and chest circumference, or measured ultrasonically.
- Note that for freshly separated U-238, the Th-234 daughter is not in equilibrium, and so lung counts are not as useful.

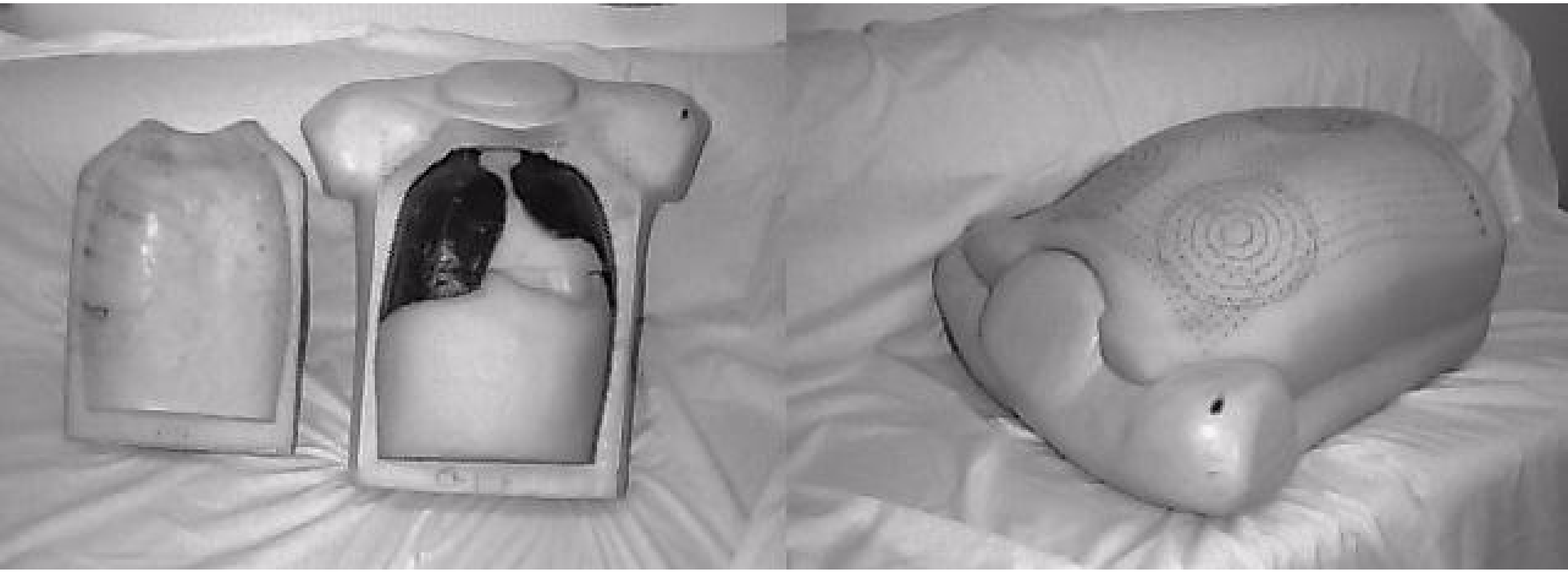



SIERRA
CANBERRA
CANBERRA

Lung Counter Calibration

- Lung counters are typically calibrated with a realistic torso phantom developed at Lawrence Livermore National Laboratory.
- Calibration is usually done on an annual basis with some form of check source used before counting.
- Long background counts are done to correct for interference from natural radionuclides.
- Typical minimal detectable activity (MDA) for lung counters are about 2 nCi for U-238 and 0.1 nCi for U-235.
- The most restrictive ALI for U is 40 nCi.

The LLNL Torso Phantom



Wound Counting

- Wound counting can be used to assess the level of uranium in a contaminated wound.
- The ratios of the low energy photons can also be used to determine the depth at which the majority of contamination is located.
- Typically a small (2.5 cm diameter) NaI(Tl) detector or an HPGe detector is used.

Typical wound counter—NaI(Tl)

Excreta Analysis

- Excreta analysis is the analysis of materials excreted by or removed from the body.
- Samples are usually urine or feces, but other body fluids can be analyzed such as saliva, perspiration, blood, and mucus (nasal swabs).
- Nail and hair clippings can be collected and analyzed as indicators of chronic intakes
- For uranium analysis urine bioassay is almost always used.

Excreta Analysis

- The preferred collection period for urine or feces is 24 hours.
- If this cannot be done, then other options are used such as the modified 24 hour collection.
  - This is collecting the last voiding before bed, any during the night, and the first void of the day for two consecutive days.
- In addition, spot samples can be used.
  - A spot sample is a small sample collecting the volume from a single void during the 24 hour period.
- If any sample other that a complete 24-hour collection is used, then a correction must be made to obtain the daily excretion.

Reference Man

- ICRP Publication 23 (1974) contains all the biological information for “Reference Man”.
- For bioassay purposes the document gives the amount of urine and feces voided during a 24 hour period for both men and women.
  - Women excrete 1 L of urine and 110 g of feces a day.
  - Men excrete 1.4 L of urine and 135 g of feces a day.
  - ICRP Publication 89 (2002) has updated this value to 1.6 L per day

Methods of Analysis

- There are four common methods of detecting uranium in urine:
  - Fluorimetry
  - Kinetic Phosphorescence Analysis (KPA)
  - Mass Spectrometry
  - Alpha Spectrometry

Fluorimetry

- Is a chemical analytical method for total uranium.
- Uranium is precipitated out of solution and fused with fluorides.
- The fluorescence of a heated sample is measured.
- The MDA for this method is about 1 µg per liter of urine.

Kinetic Phosphorescence Analysis

- KPA is similar to fluorimetry, except the phosphorescence is stimulated by a laser.
- A time window is used for the uranium phosphorescence to eliminate light from contaminants.
- The MDA is about 0.1 µg per liter of urine.
- Chemical processing is important, because organics will interfere and give a higher result.

Mass Spectrometry

- Mass Spectrometry for uranium is identical to that for other elements.
- The advantage is that chemical processing is not required before analysis.
- Typical MDA of about 0.2 µg per liter of urine.
- The disadvantage is that not many mass spectrometer owners like running radioactive samples through their devices!

Alpha Spectrometry

- The only radiometric method available for urine analysis and the only one capable of determining the enrichment of the uranium sample.
- A tracer, usually U-232, is added to the sample at the start of the analysis procedure to measure the chemical recovery.
- The uranium is separated by ion-exchange methods, and is deposited on a counting planchet.
- The typical MDA is 0.1 dpm per liter of urine.

Background Levels

- Background levels of uranium are observed in most bioassay samples due to intake of uranium in food and water.
- A typical background level of uranium in a urine sample ranges from about 0.03 to 3 µg per liter of urine.
- A facility must establish the local background level.
- In addition, most glassware contains uranium that can be leached by strong acids, so laboratories must pretreat any glassware that will be used for uranium analysis.

Frequency of Bioassay Measurements

- The appropriate frequency of bioassay measurements depends upon the exposure potential and the physical and chemical characteristics of the radioactive material.
- Elements to consider:
  - The potential exposure of the individual
  - The retention and excretion characteristics
  - Sensitivity of the measurement technique.
  - The acceptable uncertainty in the estimate of intake and dose determination.

Frequency of Bioassay Measurements

- Bioassay measurements used for demonstrating compliance with the occupational dose limits should be conducted often enough to identify and quantify potential intakes that during a year are likely to collectively exceed 0.1 ALI.
- The 0.1 ALI criterion is based on 10CFR20.1502(b) which requires licensee to monitor intakes for exposed individuals who are likely to exceed 10% of the applicable limit.

Types of Bioassay Measurements

- There are two separate categories of bioassay measurements:
  - Routine
  - Special monitoring

Routine Measurements

- Routine measurements are used to confirm that appropriate controls exist and to assess dose.
- They include:
  - Baseline Measurements
  - Periodic Measurements
  - Termination Measurements

Baseline Measurements

- An individual's baseline measurement of radioactive material within the body should be conducted prior to work activities that involve exposure to radiation or radioactive materials , for which monitoring is required.
- This measurement allows the facility to determine if a worker has had a previous intake from other facilities.
- Note that if a worker has already had an ALI, he cannot receive any external dose for the rest of the year.

Periodic Measurement

- The periodic measurement should be conducted at frequency that is determined on an *a priori* basis, considering the likely exposure to the individual.
- Periodic measurements should be made when the cumulative exposure to airborne radioactivity, since the most recent bioassay measurement, is >0.02 ALI (40 DAC-hrs).
- At a minimum, periodic measurements should be conducted annually.

Termination Measurement

- When an individual is no longer subject to the bioassay program a termination measurement should be made.
- This is to ensure that any unknown intakes are quantified.
- Can be difficult to get this sample.

Special Monitoring

- Special monitoring is used to evaluate suspected intakes of radioactive material.
- Circumstances that would indicate that special monitoring is needed are:
  - Presence of facial or nasal contamination.
  - Operational events with a reasonable likelihood that a worker was exposed to unknown quantities of radioactive material.
  - Known or suspected incidents of a worker ingesting material.
  - Incidents that resulted in skin contamination or wounds.
  - Evidence of damage or failure of respiratory protection.

Special Monitoring

Table 5-15. Uranium Levels for Internal Dosimetry Notification

Indicator	Notification Level
Nasal or mouth smears	Detectable activity
Facial contamination (direct measurement)	200 dpm
Skin breaks or blood smears	Any skin break while handling material other than sealed sources
Head, neck contamination	2,000 dpm
Contamination in respirator	Detectable activity inside respirator after use
Hands, forearms, clothing[(a)]	10,000 dpm
Airborne radioactivity	Acute intake equivalent to 40 DAC-hours after accounting for respiratory protection factor[(b)]

(a) Clothing contamination levels apply to exposure without respiratory protection, such as on inner coveralls or personal clothing.

(b)

$$\text{DAC-hours} = \frac{\text{airborne concentration}}{\text{DAC}} \times \text{hours of intake}$$

Estimating Intakes

- Intakes should be estimated for any bioassay sample that indicates internally deposited radioactive material from work activities.
- The scope of the evaluation should be commensurate with the potential magnitude of the intake.
- For individual exposures with a an estimated intake of less than 0.02 ALI, minimum bioassay measurements are adequate.
- Repeated follow-up measurements are not necessary.

Interpretation of Bioassay Measurements

- The specific scope and depth of the evaluation of bioassay measurements depends on the potential significance of the intake.
- The models that are presented are acceptable to the NRC staff for correlating bioassay measurements to estimates of intakes for the purpose of demonstrating compliance with the occupational dose limits of 10CFR20.1201.
- This is based on RegGuide 8.9 “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program” Rev 1 July 1993.

Time of Exposure

- Accurate estimation of intake from bioassay measurements depends upon knowing the time of intake.
- Generally the time of intake is determined from work activities and other monitoring data.
- When the time of intake cannot be determined from monitoring data, it can often be provided by the individual.
- When the information is insufficient to determine the time of intake it is acceptable to assume that the intake occurred at the midpoint of the time period since the last bioassay measurement.

Acceptable Biokinetic Models

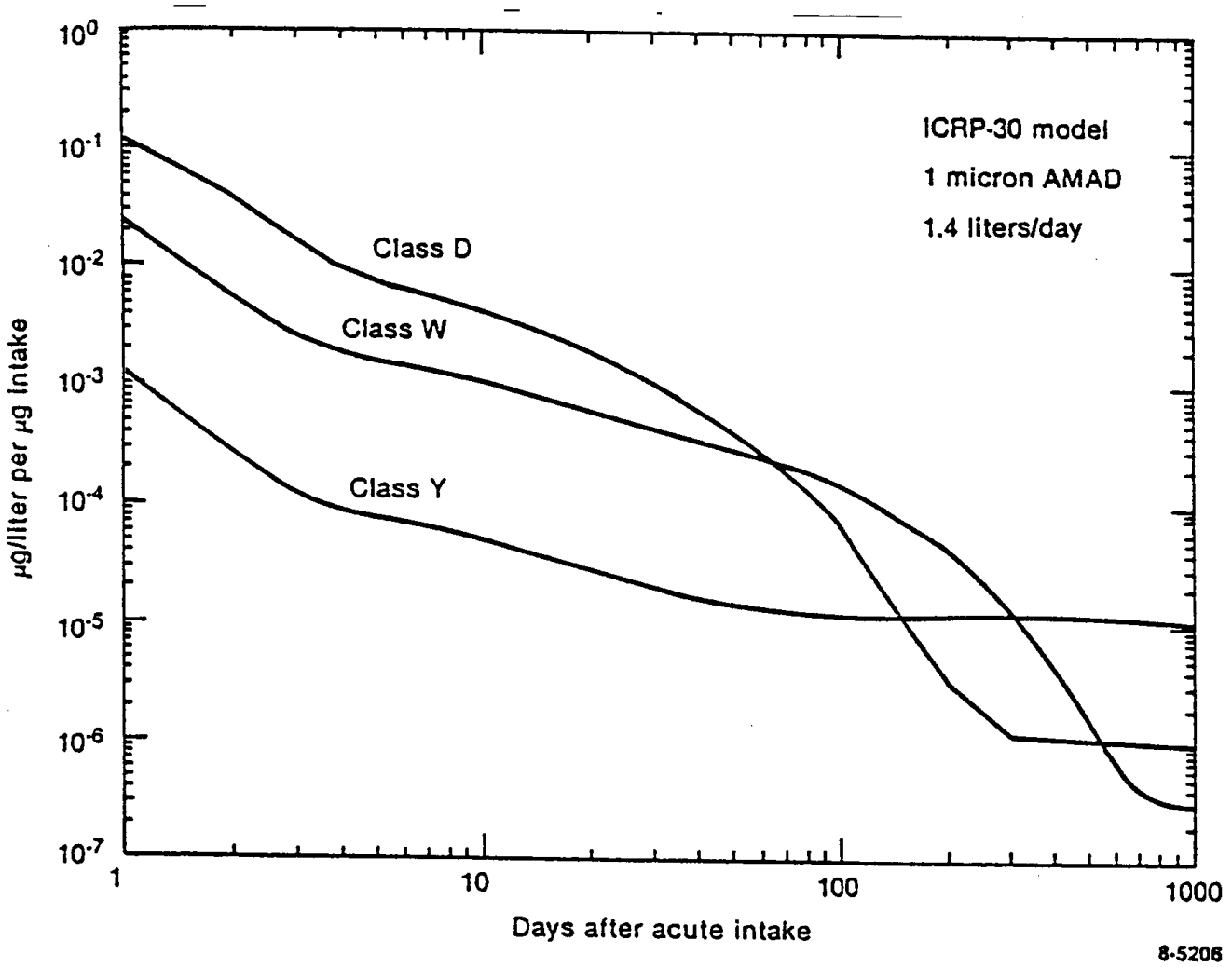
- The biokinetic models of ICRP 30 present acceptable basis for estimating intakes from bioassay measurements.
- The use of computer codes that apply these models is also acceptable for evaluating bioassay measurements provided it can be demonstrated through documented testing that the models and methods used are consistent with the acceptable models.
- Computer codes that have been used:
  - INDOS by Ken Skrable can only do one measurement method (i.e urine, fecal, or lung count)
  - Code for Internal Dosimetry (CINDY) by PNNL can use multiple measurement types and can be used for acute and chronic intakes.
  - Individual Modules for Bioassay Analysis (IMBA), newest code.

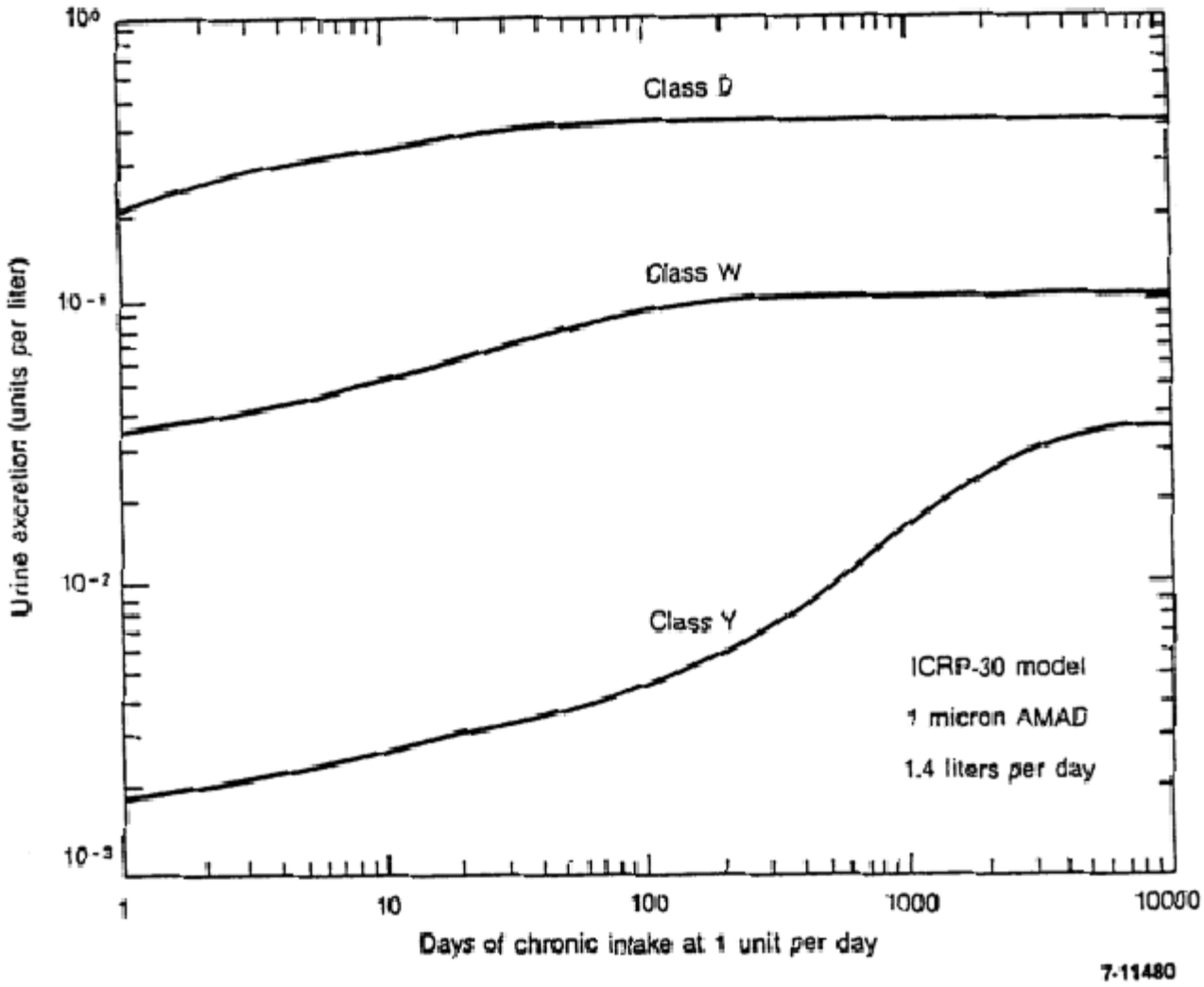
Acceptable Biokinetic Models

- Variations from the predicted retention and excretion for specific individuals is expected.
- Important considerations for bioassay evaluation are:
  - Representativeness of 24-hour urine or fecal measurements
  - The appropriate lung clearance class and particle size
  - Effect of chelation or other therapy to accelerate excretion
- Should note that experience has shown that the first voiding following exposure, while providing useful information for assessing potential significance, does not generally represent equilibrium conditions and should not be used to evaluate intakes.

Calculating Intakes

- ICRP Publication 54 presents urinary excretion and fecal excretion equations as a function of time following intake for a number of radionuclides.
- The solution to these equations over a range of times allows the development of tabulated intake retention and excretion fractions (IRF).
- The IRFs published in NUREG/CR-4884 were developed in this manner and represent an acceptable basis for deriving estimates of intake from bioassay measurements.


ICRP-30 model
1 micron AMAD
1.4 liters/day
Class D
Class W
Class Y
µg/liter per µg intake
Days after acute intake
8-5206


Urine excretion (units per liter)
Class D
Class W
Class Y
ICRP-30 model
1 micron AMAD
1.4 liters per day
Days of chronic intake at 1 unit per day
7-11480

IRF from NUREG/CR-4884

CLASS Y AMAD = 1 MICRON HALFLIFE= 1.16E+12 DAYS URANIUM 238

TIME AFTER SINGLE INTAKE	FRACTION OF INITIAL INTAKE IN:				
DAYS	SYSTEMIC ORGANS	LUNGS	NASAL PASSAGES	GI TRACT	TOTAL BODY
1.00E-01	3.54E-03	3.04E-01	2.58E-01	7.30E-02	6.39E-01
2.00E-01	3.24E-03	2.85E-01	2.17E-01	1.33E-01	6.38E-01
3.00E-01	3.02E-03	2.70E-01	1.82E-01	1.82E-01	6.37E-01
4.00E-01	2.86E-03	2.57E-01	1.53E-01	2.21E-01	6.35E-01
5.00E-01	2.73E-03	2.47E-01	1.29E-01	2.52E-01	6.31E-01
6.00E-01	2.63E-03	2.38E-01	1.08E-01	2.76E-01	6.25E-01
7.00E-01	2.54E-03	2.31E-01	9.12E-02	2.93E-01	6.18E-01
8.00E-01	2.46E-03	2.24E-01	7.67E-02	3.05E-01	6.09E-01
9.00E-01	2.40E-03	2.19E-01	6.45E-02	3.12E-01	5.97E-01
1.00E+00	2.34E-03	2.13E-01	5.42E-02	3.15E-01	5.85E-01
2.00E+00	2.02E-03	1.80E-01	9.59E-03	2.32E-01	4.24E-01
3.00E+00	1.87E-03	1.65E-01	1.70E-03	1.25E-01	2.93E-01
4.00E+00	1.76E-03	1.57E-01	3.00E-04	6.06E-02	2.19E-01

IRF from NUREG/CR-4884

CLASS Y AMAD = 1 MICRON HALFLIFE= 1.16E+12 DAYS URANIUM 238

TIME AFTER SINGLE INTAKE	FRACTION OF INITIAL INTAKE IN:			
DAYS	24-HOUR URINE	ACCUMULATED URINE	24-HOUR FECES	ACCUMULATED FECES
1.00E-01		3.80E-04		1.04E-05
2.00E-01		7.49E-04		2.10E-04
3.00E-01		1.05E-03		1.06E-03
4.00E-01		1.31E-03		3.07E-03
5.00E-01		1.53E-03		6.66E-03
6.00E-01		1.71E-03		1.21E-02
7.00E-01		1.87E-03		1.94E-02
8.00E-01		2.01E-03		2.86E-02
9.00E-01		2.13E-03		3.95E-02
1.00E+00	2.23E-03	2.23E-03	5.20E-02	5.20E-02
2.00E+00	5.49E-04	2.78E-03	1.60E-01	2.13E-01
3.00E+00	2.30E-04	3.01E-03	1.31E-01	3.43E-01
4.00E+00	1.57E-04	3.17E-03	7.35E-02	4.17E-01
5.00E+00	1.31E-04	3.30E-03	3.63E-02	4.53E-01
6.00E+00	1.17E-04	3.41E-03	1.72E-02	4.70E-01
7.00E+00	1.07E-04	3.52E-03	8.11E-03	4.78E-01
8.00E+00	9.81E-05	3.62E-03	3.88E-03	4.82E-01
9.00E+00	9.07E-05	3.71E-03	1.91E-03	4.84E-01
1.00E+01	8.42E-05	3.79E-03	9.83E-04	4.85E-01

Calculation of Intake

- The intake is calculated using the following equation:

$$I = A / IRF$$

Where I = estimate of the intake with units the same as A,

A = numerical value of the bioassay measurement with appropriate units,

IRF = intake retention fraction corresponding to the type of measurement, intake route, and solubility class.

Example

- A single bioassay measurement from a 24 hour urine sample indicates an activity of 1 pCi U-238. An investigation indicates that the exposure occurred 3 days before the sample was taken, and the compound was class Y. What is the estimated intake?

$$I = A / IRF$$

A = 1 pCi

IRF for 3 days post exposure for class Y U-238 = 2.3 E-4

$$I = 1 \text{ pCi} / 2.3E\text{-}4 = 4347 \text{ pCi or } 4.3 \text{ nCi}$$

Dose Calculation

- To calculate the dose the dose conversion factors or the ALI can be used.
- If the ALI is used the calculation is D = (I / SALI) * 5 rem

  or D = (I / NALI) * 50 rem
- In this example, it is class Y, so the SALI would apply and

  D = (4.3 nCi / 40 nCi) * 5 rem = 0.538 rem
- If the DCF is used the equation is D = I * DCF (mind the units)
- So in this example:

  D = 4.3 E-9 Ci * 3.2 E-5 Sv/Bq * 3.7 E12 (rem-Bq)/(Sv-Ci) = 0.510 rem

Using Multiple Measurements to Calculate Intake

- When multiple bioassay measurements are made a statistical evaluation of the data should be performed.
- NUREG/CR-4884 recommends the use of an unweighted least squares fit.
- The equation is as follows:

$$I = \frac{\sum IRF(t) \times A\,(t)}{\sum IRF(t)^2}$$

Example

- A person is working in an area that processes class Y U-238 in an enclosed hood. During work, a CAM at the other end of the room begins to alarm. The employee secures the work station then exits the area. 24-hour urine samples are requested on days 1, 3, and 5; the following are the results of those analysis. Using the ICRP 30 models, what is the dose to the employee?

Days after exposure	Urine bioassay result (pCi)	IRF
1	19	2.23 E-3
3	2	2.30 E-4
5	1	1.31 E-4

Example

- Intake is equal to :

$$I = \frac{\sum IRF(t) \times A\,(t)}{\sum IRF(t)^2}$$

So:

$$I = \frac{(2.23E-3\ \times 19) + (2.30E-4\ \times 2) + (1.31E-4\ \times 1)}{(2.23E-3)^2 + (2.30E-4)^2 + (1.31E-4)^2} = 8.5\ nCi$$

Dose is equal to : D = (I / ALI)* 5 rem

= (8.5 nCi / 40 nCi)* 5 rem = 1.1 rem

Evaluating Spot Samples

- If the total urine and feces for the 24 hour period is not collected the following equations may be used to estimate the total activity excreted or eliminated over the 24 hour period based on less frequent sampling.

$$\Delta A_i = C_i \times E \times (t_i - t_{i-1})$$

$$A_i = \sum \Delta A_i$$

Evaluating Spot Samples

- ΔA_i = Activity or amount of radioactive material in sample i
- i = The sequence number of the sample
- C_i = The radionuclide concentration in urine (activity / L) or feces (activity / g) of sample i
- E = Daily excretion rate
- t_i = The time (days) after the intake that sample i is collected
- A_i = Total activity excreted or eliminated up to time t_i

Evaluating Spot Samples

- In general spot samples should be collected frequently enough that there is not more than a 30% change in the IRF between bioassay measurements.
- If the IRF changes by 30% a day then a daily spot sample is required.
- If the IRF changes 10% a day then a spot sample would be done at least every three days.
- The rapid clearance and excretion of inhaled particles from the N-P region of the lung makes it important that at least one spot sample be collected in the first 24 hours.

Adjusting for Multiple and Continuous Intakes

- In practice a worker may receive repeated exposure to the same radionuclide over a period of time.
- These intakes should be treated as separate acute intakes if measurements collected through the period allow for individual quantification of each exposure.
- If the intakes are separated in time so that the retained or eliminated fraction from an earlier intake is less than 10% of the retention or excretion fraction for the next intake, the intakes can be evaluated separately without regard to any previous intakes.

Adjusting for Multiple and Continuous Intakes

- Continual intakes that are distributed equally in size and time may be approximated using a relationship based on time integration of the IRF.
- The total intake is estimated by dividing the measured activity by the appropriate time-integrated retention or excretion fraction.
- Needless to say this is complex and is best accomplished using computer software.

Correcting Intake Estimates for Particle Size Differences

- The models used for deriving IRFs are based on a 1 micron activity mean aerodynamic diameter (AMAD) particles.
- It is acceptable to correct intake estimates for particles of different sizes.
- These corrections often help explain retention or excretion rates that are different from those expected, such as would occur for larger particles preferentially deposited in the N-P region with more rapid clearance times.

Correcting Intake Estimates for Particle Size Differences

- The equation given may not provide valid corrections for time periods shortly following intakes.
- The time after intake for which the equation yields satisfactory results is less than 1 day for class D, 7 days for class W, and 9 days for class Y.
- ICRP 30 provides data and methods for use in doses for particle sizes between 0.1 to 20 μm AMAD.
- For particles with AMAD greater than 20 μm complete deposition in the N-P region can be assumed.

Regional Deposition Fractions for Aerosol with AMAD between 0.2 and 10 um

	0.2 μm	0.5 μm	0.7 μm	1.0 μm
D_{N-P}	0.05	0.16	0.23	0.30
D_{T-B}	0.08	0.08	0.08	0.08
D_P	0.50	0.35	0.30	0.25
Total Deposition	0.63	0.59	0.61	0.63
	2.0 μm	5.0 μm	7.0 μm	10.0 μm
D_{N-P}	0.50	0.74	0.81	0.87
D_{T-B}	0.08	0.08	0.08	0.08
D_P	0.17	0.09	0.07	0.05
Total Deposition	0.75	0.91	0.96	1.00

Correcting Intake Estimates for Particle Size Differences

- The equation below is taken from appendix B of NUREG/CR-4884 and should be used for revising the total body IRFs for particle size distributions from 0.1 to 20 µm AMAD.

$$IRF_{(AMAD)} = IRF(1\mu m)\sum_T \left[f_{N-P,T} \frac{H_{50T}W_T}{\sum_T H_{50T}W_T} \frac{D_{N-P}(AMAD)}{D_{N-P}(1\mu m)} + f_{T-B,T} \frac{H_{50T}W_T}{\sum_T H_{50T}W_T} \frac{D_{T-B}(AMAD)}{D_{T-B}(1\mu m)} + f_{P,T} \frac{H_{50T}W_T}{\sum_T H_{50T}W_T} \frac{D_P(AMAD)}{D_P(1\mu m)} \right]$$

Correcting Intake Estimates for Particle Size Differences

- It is acceptable to compare the estimate of intake for different particle sizes with the ALIs in Appendix B of 10CFR20 for demonstrating compliance with the intake limits.
- The ALIs are based on a 1 um AMAD particle size.
- Modifying the ALI values for different particle size distributions requires prior NRC approval (10 CFR 20.1204(c)(2)).

Use of Individual Specific Models

- Individual-specific IRFs may be used in developing biokinetic models that differ from the standard models.
- The quality and quantity of the data used for this type of individual-specific modeling should be sufficient to justify the revised model.
- Licensees should not attempt to develop individual-specific models in the absence of actual biochemical and particle size information.

Evaluation Level

- For very small intakes a single bioassay measurement is generally adequate to estimate intake.
- For intakes that represent a significant contribution to dose, other available data should be evaluated.
- If the estimate indicates an intake greater than 0.02 ALI additional data, such as airborne radioactivity measurements or additional bioassay measurements, should be used to obtain the best estimate of actual intake.

Investigation Level

- For single intakes that are greater than 10% of the ALI a thorough investigation of the exposure should be made.
- If the investigation level is exceeded, multiple bioassay measurements and an evaluation of all available monitoring data should be conducted.
- If practical, daily bioassay measurements should be conducted until a pattern of bodily retention and excretion can be established.
- Usually this pattern can be established after three measurements, however, due to uncertainties in physiology, measurements should be conducted over a longer period of time.

What to Look for in an Internal Dosimetry Program

- When evaluating a program for determining internal dose some items to look for are:
  - Policy requiring participation in bioassay program by appropriate workers.
  - Implementing procedures (scheduling, sample kit instructions, sample kit issue / receipt, data handling).
  - Arrangements with appropriate analytical labs, including specifications of analysis, sensitivity, processing times, reporting requirements, and quality assurance provisions.
  - Onsite support facilities (e.g. sample kit storage locations, sample kit issue and collection stations)

What to Look for in an Internal Dosimetry Program

- Recommendations for the testing criteria of radiobioassay labs are listed in ANSI/HPS N13.30 “Performance Criteria for Radiobioassay” May 1996.
- If dealing with soluble uranium, need to be able to demonstrate compliance with the kidney toxicity limit in 10 CFR 20.
- Will have determined the sampling frequency based on missed dose concept.
- Will have determined levels at which investigation and additional sampling will be done.
- Good idea to have a restriction level above which personnel will not be allowed to work while dose is determined.

What to Look for in an Internal Dosimetry Program

- Will have determined the nuclide mix and solubility class for the uranium materials being handled.
- Most of this information should be compiled in a Technical Basis Document for internal dosimetry.
  - Facility description
  - Materials
  - Analysis methods
  - Calculational methods
  - Computer codes if used how validated.

Peeking into the Future

- Newer ICRP publications:
  - Nomenclature
  - Dose limits
  - Tissue weighting factors
  - Biokinetic models
- Coming soon to a 10 CFR 20 near you??

Changes in Nomenclature

- "Quality factor" renamed "radiation weighting factor", w_R
- "Dose equivalent" renamed "Equivalent dose"

  = absorbed dose x radiation weighting factor:

  $H_T = \Sigma_R w_R D_{T,R}$
- "Effective dose equivalent" renamed "Effective dose"

  = equivalent dose x w_T summed over tissues:

  $E = \Sigma_T w_T H_T$
- "Non-stochastic" renamed "deterministic"
- "Genetic" renamed "hereditary"
- Committed and collective remain same

ICRP Tissue Weighting Factors

Organ	ICRP 30	ICRP 60	ICRP 103
Gonads	0.25	0.20	0.08
Breast	0.15	0.05	0.12
Red marrow	0.12	0.12	0.12
Lung	0.12	0.12	0.12
Thyroid	0.03	0.05	0.04
Bone surface	0.03	0.01	0.01
Other organs	0.30 (6)	0.025 (2)	0.12 (avg)
Explicit factors	6	12	14

Annual Maximum Dose Limit

Effective dose should not exceed 50 mSv (5 rem) in any one year

This is the sum of external penetrating deep dose equivalent (DDE) and committed effective dose from all intakes in a year

Annual Average Dose Limit

- Effective dose should not exceed 100 mSv (10 rem) in a designated 5-year period
- Thus, effective dose should average 20 mSv (2 rem) per year
- The ALI's and DAC's published by ICRP and IAEA are now based on this average (20 mSv per year)
- Separate limits for lens of eye: 150 mSv (15 rem) and for skin and extremities: 500 mSv (50 rem)
- Note the limit on TODE is gone!

New Human Respiratory Tract Model

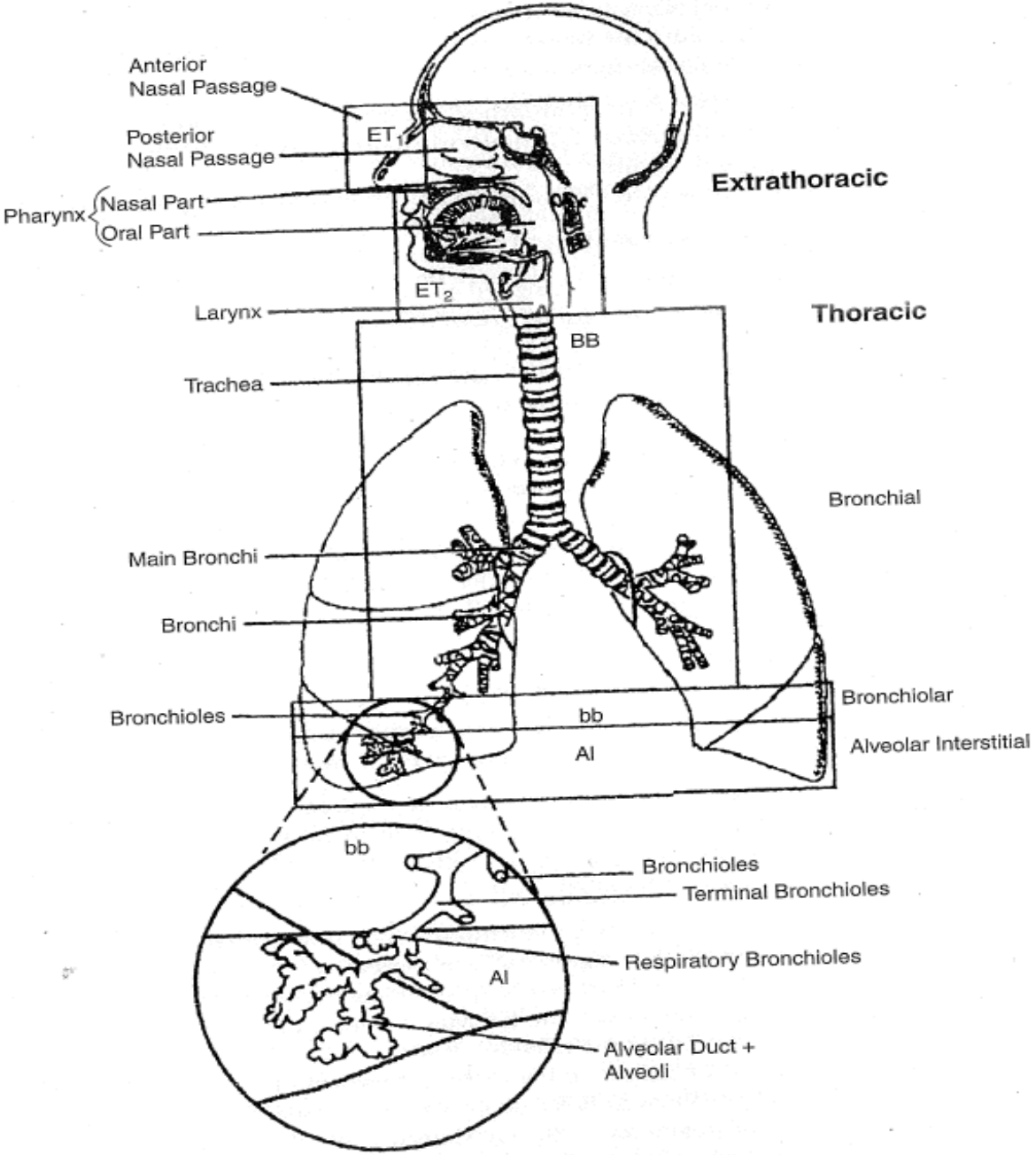
- ICRP 66 was published in 1994.
- This provides a whole new dosimetric model of the lung.
- This model resulted from increased knowledge of the biokinetics of the respiratory process, including the inhalation of aerosols and gases, the radiosensitivity of the several different tissues within the respiratory tract, and the biological effects of inhaled radioactivity.
- The default particle size is now 5 µm AMAD instead of the 1 µm AMAD of ICRP 30

New ICRP Models

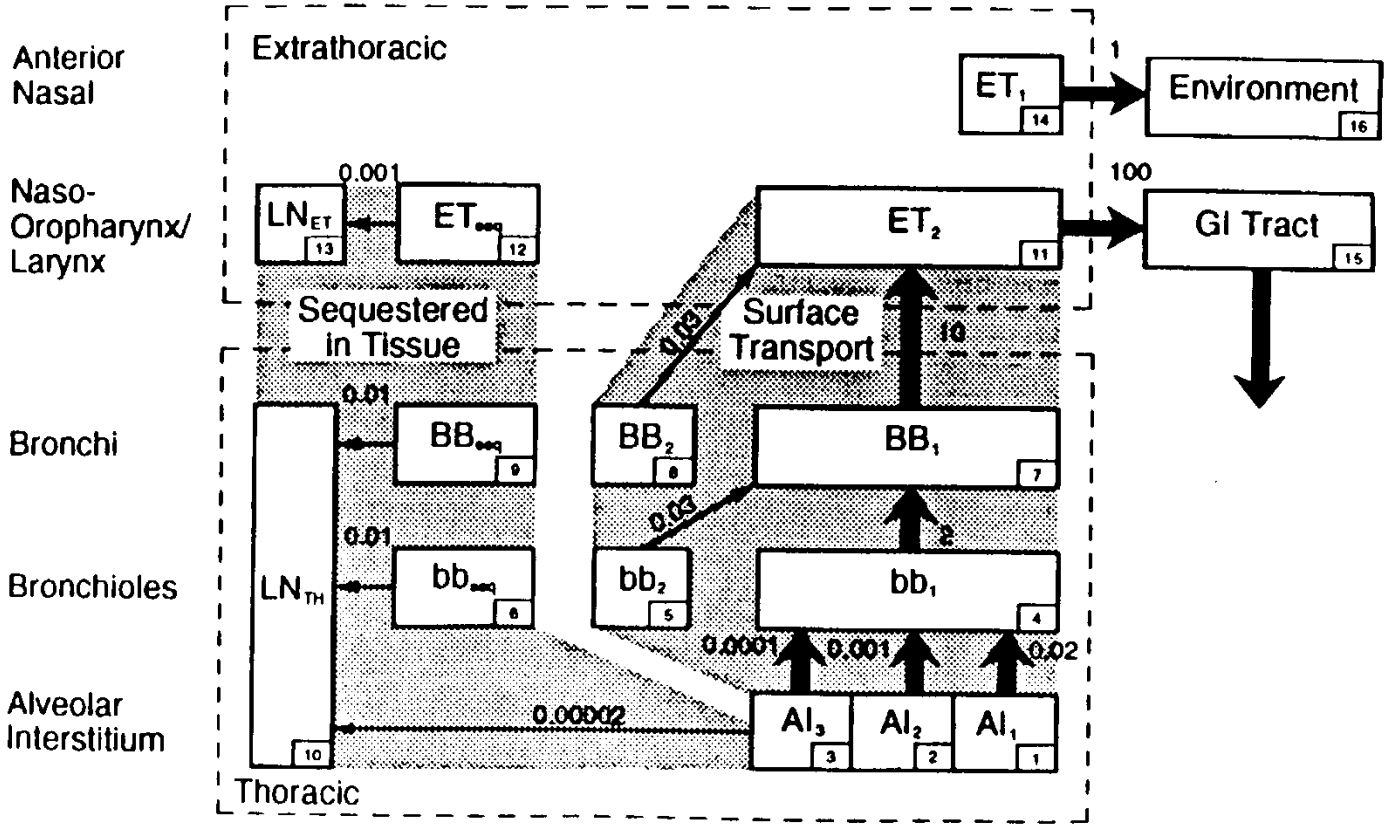
- ICRP 66 also changes the classification of particles from solubility to rate of absorption of the inhaled radioactivity into the blood.
  - Type F (fast) – 100% absorbed into blood in < 10 minutes.
  - Type M (moderate) – 10% absorbed into blood in < 10 minutes, 90% absorbed in < 140 days.
  - Type S (slow) – 0.1% absorbed in < 10 minutes, 99.9% absorbed in > 140 days.

ICRP 66 HRTM

- Extrathoracic (ET) region – the portion of the respiratory tract outside of the chest which contains two subparts:
  - ET_1 consisting of the anterior nasal airways
  - ET_2 consisting of the posterior nasal airways, pharynx, and larynx
- Bronchial (BB) region which includes the trachea and bronchi
- Bronchiolar (bb) region consisting of bronchioles and terminal bronchioles.
- Alveolar-interstitial (AI) region which consists of the respiratory bronchioles, the alveoli, and interstitial connective tissue.


Anterior
Nasal Passage
Posterior
Nasal Passage
Pharynx
Nasal Part
Oral Part
Larynx
Trachea
Main Bronchi
Bronchi
Bronchioles
ET1
ET2
BB
bb
Al
Extrathoracic
Thoracic
Bronchial
Bronchiolar
Alveolar Interstitial
Bronchioles
Terminal Bronchioles
Respiratory Bronchioles
Alveolar Duct +
Alveoli

MODEL OF TIME-DEPENDENT PARTICLE TRANSPORT
1993 ICRP MODEL



Use of the ICRP 66 HRTM

- Currently NRC regulations are based on the ICRP 30 model.
- The United States Department of Energy in 10 CFR 835 is in the process of adopting the ICRP 66 model for internal dose. (Effective 2010).
- The DCFs for the ICRP 66 model are published in ICRP 68.
- The November 2002 issue of the Health Physics Society Journal contains an article the presents the IRFs for the ICRP 66 model.
- Currently NRC licensees can request an amendment to use the ICRP 66 lung model with the ICRP 68 DCFs; all fuel fabrication facilities have this license amendment.

Structure of the Human Alimentary Tract (HAT) Model – ICRP 100

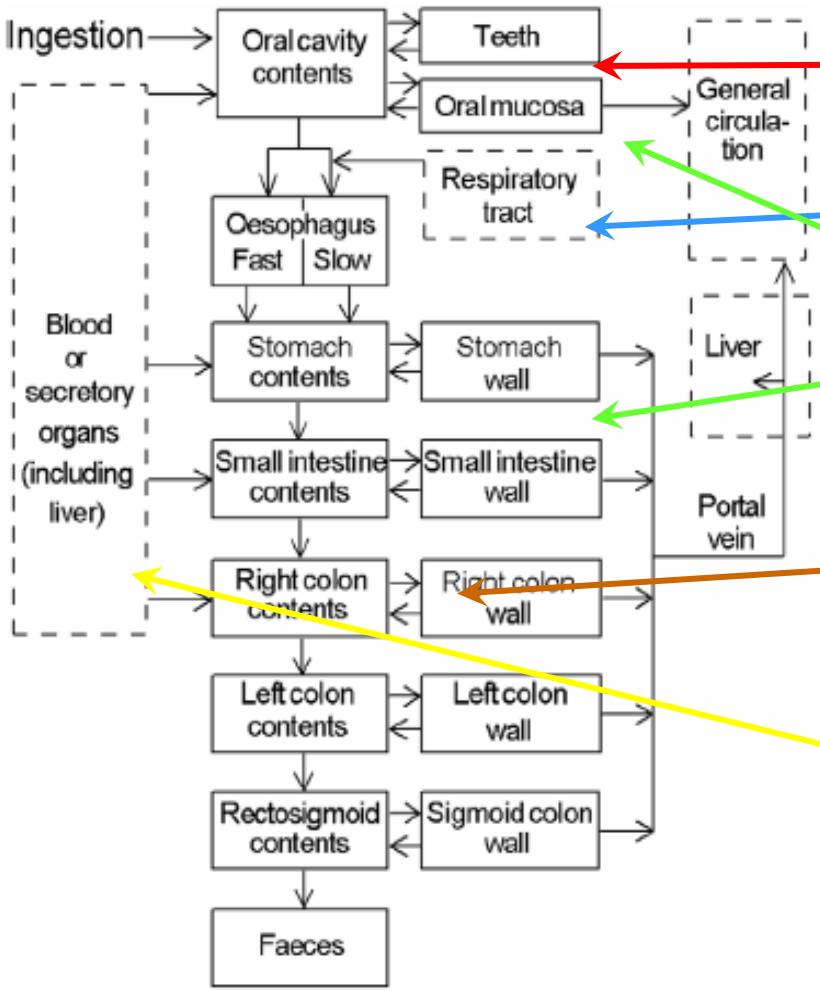
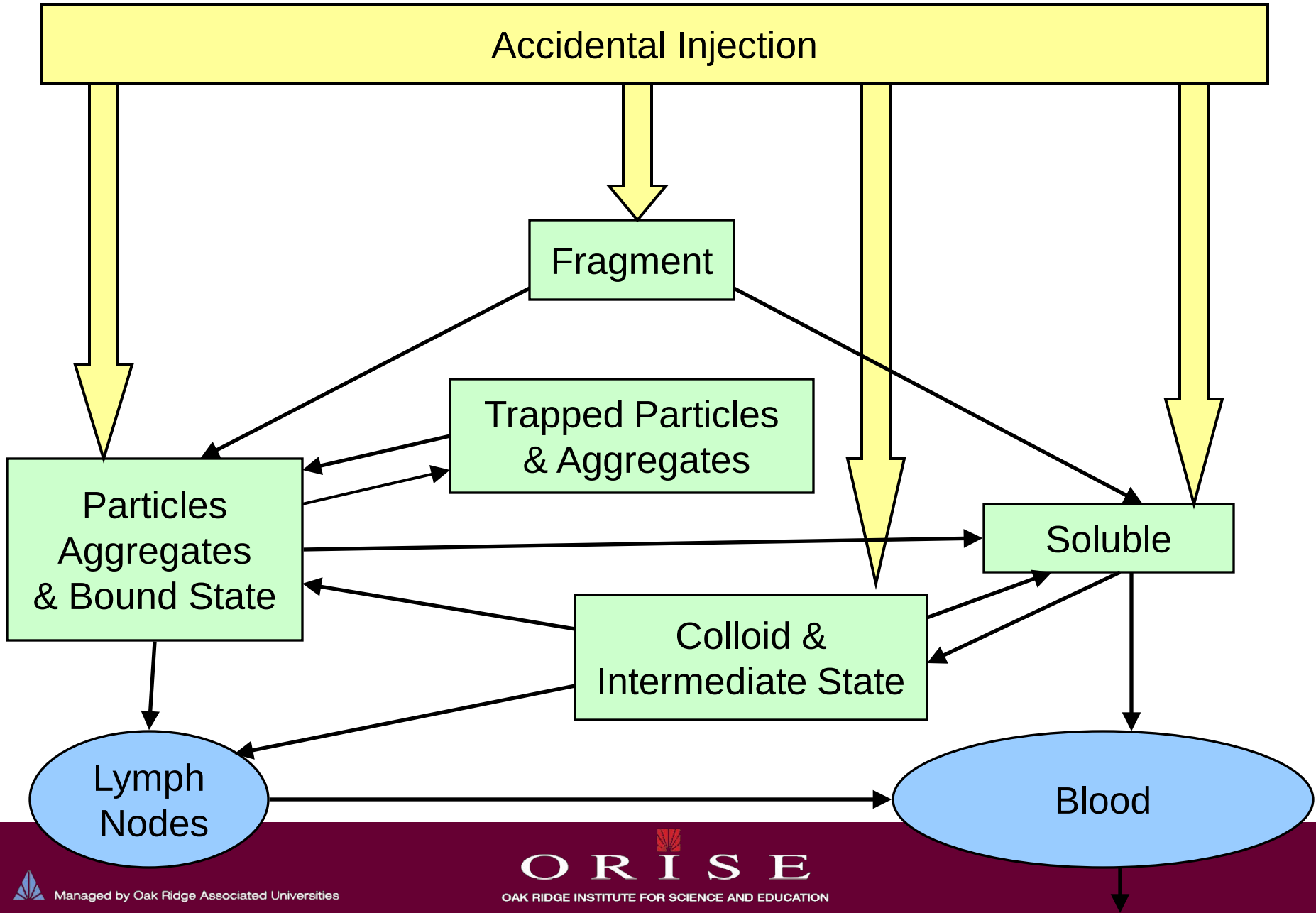


Fig. 5.1. Structure of the human alimentary tract model (HATM). The dashed boxes are included to show connections between the HATM and the human respiratory tract model or systemic biokinetic models.

- Deposition and retention on teeth.
- Entry per ingestion, from Respiratory Tract
- Deposition in oral mucosa or wall of the Stomach and intestine
- Transfer back from the oral mucosa or walls of ST and intestine back in the lumen
- Transfer from various secretory organs or blood into the contents of certain segments of HAT.

THE NCRP WOUND MODEL



Confessions of an Internal Dosimetrist

- Lessons learned from operating an internal dosimetry program for uranium.
  - If U is class Y most bioassay methods are not sensitive enough.
  - Beware of collecting data to feel good make sure it means something.
  - Air sampling is your friend.
  - Expect the unexpected.
  - This actually does work …..sometimes.
- The best thing about internal dosimetry is that no one can ever prove you were wrong!